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in co-operation with the Electric Utility Cost Group Inc.,
International Science and Technology Centre,
World Energy Council and
World Nuclear Association

IAEA

Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power

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FOREWORD

Nuclear power is a significant contributor to the global supply of electricity, and continues to be the major source that can provide electricity on a large scale with a comparatively minimal impact on the environment. But it is evident that, despite decades of experience with this technology, nuclear power today remains mainly in a holding position, with its future somewhat uncertain primarily due to concerns related to waste, safety and security. One of the most important factors that would influence future nuclear growth is the innovation in reactor and fuel cycle technologies to successfully maximize the benefits of nuclear power while minimizing the associated concerns.

The International Conference on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power, organized in June 2003 by the IAEA in co-operation with the World Nuclear Association, the World Energy Council, the International Science and Technology Center and the Electric Utilities Cost Group, comes at a pivotal point in the history of nuclear energy. Progress has been made recently on many fronts related to nuclear power, including waste disposal, license extension, and safety and security upgrades. Many countries are engaged in innovation projects and while most of the current expansion in nuclear energy is taking place in East and South Asia, recent years have witnessed statements and actions in North America, Europe and elsewhere that support a renewed consideration of the merits of nuclear power.

The conference included talks on specific topics by 21 invited speakers drawn from 11 Member States as well as 21 oral presentations and 26 poster presentations of accepted papers. Part of the opening session and two half-day sessions were devoted to panel discussions in which 23 panelists from 9 Member States and 5 international organizations took part. All relevant aspects of innovative technologies for nuclear fuel cycles and nuclear power were discussed in an open, frank and objective manner. These proceedings contain a summary of the results of the conference, invited and contributed papers, and summaries of panel discussions.

The IAEA wishes to thank the invited speakers, panelists, authors, programme committee members and all the participants for their contributions in making this conference a success. The IAEA officers responsible for this publication were V.M.R. Koorapaty and J. Kupitz of the Division of Nuclear Power and F. Fukuda of the Division of Nuclear Fuel Cycle and Waste Technology.

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POSTER PRESENTATIONS

TECHNICAL ASPECTS OF INNOVATIVE NUCLEAR SYSTEMS INCLUDING RELIABILITY AND SAFETY

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Abstract. The current stage of innovative technology selection requires extensive interaction between the regulators and engineers, designers, environmentalists, investors and politicians in order to develop mutually acceptable underlying principles, approaches and, if possible, unified standards. New regulatory standards should ensure that safety enhancements precede the increase in a scale of nuclear power development. Preliminary analysis shows that the new technology - the use of solid coolant in nuclear reactors - could meet broad range of User Requirements for innovative technologies. It seems that development of innovative technologies requires a concerted effort of the world community.

1. INTRODUCTION

Accidents at Three-Mile Iceland and Chernobyl have shown the insufficient level of nuclear power safety.

Two important thesis's were put forward:

- nuclear reactors of the future must differ in principle from today's reactors (Alvin Weinberg);
- nuclear technology shall be the "forgiving" one, i.e. a single operator error should make severe accident consequences highly unlikely.

However, designs that were produced or developed in a next 20 years were not characterized by the principle modifications. The safety enhancements were achieved by increasing the number of safety systems that, correspondingly, makes the nuclear facilities more expensive. It has and still is making nuclear power less competitive.

The decrease of nuclear power competitiveness, public concern with regard to nuclear accidents, the lack of acceptable technologies for nuclear spent fuel and waste management resulted in decrease of public confidence followed by decisions taking to reduce the nuclear power use in Sweden and Germany and slowing down in deployment of nuclear power in other countries.

The willing to find a way out of such a situation is clearly defined by Mr. M. El Baradei, the present IAEA Director General: "In my view, a solution to this dilemma may depend heavily on the development of new, innovative reactors and fuel cycle technologies. To be successful, the new technology must be absolutely safe, proliferation resistant and economically competitive."

Recognizing the need to make radical solutions, the IAEA came to a decision to develop the "User Requirements". The final first version of those Requirements became available in May of this year.

"User Requirements" mean that innovative nuclear reactors and their fuel cycles shall meet the advanced standards related to safety, environment protection and radioactive waste management and to be more cost-effective.

The "User Requirements", however, do not set forth any safety conditions that could be interpreted as being more stringent than those contained in the best examples of currently existing reactor and nuclear fuel cycle technologies.

2. REGULATOR'S ROLE IN SEARCH OF INNOVATIONS

The word "User" in the IAEA terminology has a broad meaning. It includes investors, designers\engineers, utilities, Regulatory Bodies, and consumers of nuclear power industry products. Therefore, one can see the great divergence of interests among the members of this community.

Formulation of "User Requirements" will facilitate broad-based international consensus among different User categories regarding acceptance of new technologies, enabling the Users to compare them with each other as well as with the existing technologies.

The Licensing Authorities play specific role as spokesman of combined requirements of community, therefore they might become a key persons in search of optimum correlation for all requirements and criteria.

The representatives of Licensing Authorities together with designers, engineers, fabricators, economists and environmentalists should participate in discussions of innovative technology proposals from the very start. It will allow them to make their recommendations at the benefit of those technologies that should be further developed.

It seems reasonable to discuss two topics, namely:

- International co-operation of Licensing Authorities that might facilitate improvement of "licensing infrastructure" and licensing activity, and
- Development of mutually acceptable underlying principles, approaches and, if possible, unified standards to be applied to innovative technologies at the international level.

The developing countries where nuclear power will be developing at a faster than average pace could become directly involved in this collaboration.

Regulatory Authorities are responsible for formulation, approval and entering into force of Regulatory Requirements.

3. THE INCREASING SCALE OF NUCLEAR POWER INDUSTRY CALLS FOR CORRECTION OF REGULATIONS

Regulatory Requirements change with the development of society and technology. Predicted deployment of nuclear energy application shall not be accompanied by increase of probability of radiological impact. Requirements related to significant radioactive release probability should be tightened at a faster rate than the rate of growth in the nuclear industry.

Since in reality the nuclear installations of the old and innovative design will have to coexist for a rather long time, this increased stringency will naturally have a greater impact on innovative projects.

Such logic leads to the necessity to assume as the innovative design only those reactors in which the considerable radioactive releases are intrinsically impossible and brought to negligibly small amount while considering all force-majeure external circumstances.

The same requirements relate to the external fuel cycle as well, assuming also that in this case one should consider not only the emergency, but also the routine releases\discharges, mainly to the air and water. The release of long-lived Alpha-sources such as plutonium shall be considerably reduced taking into account the development of radiochemistry and cumulative effect of discharges..

Review of the recently received innovative technology proposals demonstrates once again the complexity of generating truly innovative ideas. A lot of interesting and useful proposals available represent the logical development of existing technologies that satisfactory proved from the today's position. The more practical experience there is in the use of economic and safety indicators, the more confidence inspire.

This logic points us toward the evolutionary path of nuclear power development. The "Guidelines of INPRO" provides the reactor types that characterized by the wide scope of experience of use (such as reactors with light water) or, at least, by the long development history (as the high-temperature reactors and breeders with liquid-metal coolant).

These Projects have the advantage of better understanding of processes and the higher degree of assessment reliability, however, unfortunately, they have also the predetermined limits of economic and safety indicator improvement.

We appreciate very much the "principles" and "requirements" described in the "INPRO" but assume that two matters should be emphasized. For the broad development of nuclear power industry, besides of reproduction of nuclear fuel, it is necessary to use the widespread natural materials only. Existing technologies of thermal reactors lead to irretrievable spending of, mainly, zirconium and the attempts of wide use of breeders on the modern technological basis shall lead to increased irretrievable spending of such materials as chrome, nickel, manganese, tungsten. This fact is important for the future attempt of transition to use of thermonuclear reactors based on the electromagnetic confinement and also it might become substantial for the innovative technologies as well while assuming considerable increase in scale of nuclear power industry especially with regard to breeders with liquid-metal coolant.

Assuming that the number of reactors in the world increases by several-fold, transition to reprocessing of most of spent-fuel presents a problem in its own right.

Existing regulations covering discharges of alpha-active product by nuclear fuel cycle facilities will be revised that will prompt changes in relevant requirements.

Future increase in the thermal reactor fuel burn-up fraction due to an enhanced neutron economy may eventually help solve this problem. The increase of conversion factor in the reactor up to, for instance, the 0.85 (that is achievable in case of good neutron-physical characteristics in the thermal reactors) reduce the need of natural Uranium by two-to-threefold. This makes it possible to speak about partially closing the nuclear fuel cycle in the power reactor core.

In case of having a good neutron balance and the possibility to organize utilization of neutron leakage from the reactor core it would be possible to use a part of nuclear material for conversion of fissionable nuclides. In doing so, a part of nuclear material will not be a subject for enrichment or will not contain the (radiochemical) reprocessing products. To implement this idea the easily accessible free space available closely to and inside the reactor core is necessary and this should be combined with good neutron economy within the reactor core.

Such an approach requires increasing a number of fuel transpositions within and out of the reactor core and, hence, the nuclear reactor construction shall be adjusted for this. A part of fuel with high burn-up fraction will not need to be reprocessed.

4. INNOVATIVE PROJECTS REQUIRE TRUE NOVELTY

Use of well-known technologies with light and heavy water as well as of fast breeder reactors with liquid-metal coolant has got the limited capabilities and, probably, cannot resolve the contradiction, namely: to make a technology much safer and, at the same time, more economically competitiveness with the conventional power industry.

The solution to this problem could be found only by creating principally new technologies. In case of finding such technologies they will need to be supported so as to obtain evidence of their high performances within the restricted time.

It would, probably, be difficult to receive such support because for a long time key companies have invested their resources into the development of conventional technologies and, therefore, have a vested interest in pursuing traditional activities.

It is appropriate to recall that, practically, all currently available and well-developed technologies were produced while being supported by heavy government subsidies. There were military and later on – the civil programs.

It should be recognized that the really innovative technologies would require joint efforts of many countries.

5. SOLID COOLANT AS AN OPTION OF INNOVATIVE APPROACH

Study of information provided upon the Generation IV Program and the "INPRO" Requirements have led the RF Gosatomnadzor' Management to the idea that it would be expedient to evaluate the still restricted investigation results obtained in the Tomsk-7 (now the Seversk, Russian Federation) collected starting from the end of 1980th and continued since 1994 at the "Luch" Enterprise (Podolsk, Moscow Region, Russia). The case in point is the materials justifying the use of solid coolant in nuclear reactors.

What makes this option attractive is the impression that the above dilemma could be resolved, to wit: to make a technology much safer, more environment-friendly and cheaper. One of the important features of solid coolant (if the material is able to withstand high temperatures as a solid one) is the absence of phase transient. This property makes the solid coolant rather simple and reliable from the point of view of prediction of its behavior in normal and emergency conditions. It also serves as a very good addition to the main properties of solid coolant, namely, to the possibility to have a pressure that is lower than the atmospheric one in conditions of very high temperature. The low pressure in combination with properties of solid coolant, in case of correct choice of constructive decisions, will mean the rather low probability of emergency processes and negligence of their consequences.

Materials with law capture cross-section that remain solid in the high temperature conditions do exist in the nature. The first but not only one of them is the graphite. It has a low capture cross-section, the very high sublimation temperature, up to a certain high temperature point its strength increases with temperature. Pyrolytic graphite does not support natural combustion up to temperatures above 1000 C. This is why it is widely used, in particular, in the missile technology. This element is widespread in nature. Pure graphite, practically, cannot be activated by the neutrons and has got negligible induced activity. Under such conditions it becomes clear why studies of solid coolant was started with graphite. Besides, the graphite has got a high thermal capacity, therefore the Reactor with thermal power of 3500 MW with an inlet coolant temperature of 450 C and outlet temperature of 850 C within the reactor core shall have the flow rate of 5 ton per second only.

At present the following might be considered as having been established with high degree of confidence:

- Optimal size and form of particles are defined;
- Conditions for organizing of reliable continuous movement of solid coolant via the Nuclear Reactor Core are formulated;
- The capability to organize continuous flow is experimentally proved;
- Flow velocity for continuous flow is defined;
- Density variations of solid coolant during its movement through the round channel are determined.

Of course, the values of acceptable flow velocity for solid coolant is considerably law than those for the gas and water coolant – it comes to dozens of centimeter per second.

Since available data related to flow velocity for solid coolant of chemical reactors, as a rule, are considerably law rather than those in nuclear reactors, and because of lack of data available in the literature with regard to heat transfer factor for the materials based on carbon selected as a coolant, for the flow velocity of 0.1 – 0.3 meter per second (that is acceptable for nuclear reactor), the necessity was recognized to build special installations and perform some experiments with the aim to obtain the values of heat transfer factors.

During the years of 1996-1999 the data related to heat transfer factors were obtained with acceptable estimated error. It was done for different types of gas medium in the experimental volume of argon, helium and vacuum.

Experimental studies of solid coolant wear under thermal cycling were started. By the end of 2002 the test duration reached 1000 hours during which the coolant was subjected to approximately 120 thousand cycles of passing through the heating part (that simulated the reactor core) and cooling zone (simulating the secondary heat exchanger).

These experiments demonstrated that based on the extent of coolant wear, one could predict its lifecycle as being on the order of 8-10 thousand hours.

Core neutronics analyses were performed for various solid coolant configurations. These calculations made it possible to select an acceptable graphite-to-uranium ratio in the core that meet the operational requirements and likely accident conditions.

6. WHAT DO WE GET BY USING SOLID COOLANT?

Since a reactor with this type of coolant is expected to operate in the high-temperature region for the purpose of utilizing high-temperature heat or for increasing its electrical efficiency and lowing the amount of waste heat, the pyrolytic carbon-coated particles are being considered now as fuel, similar to high-temperature gas-cooled reactor fuel.

During the experiments the temperature of heating wall was brought to 800 C and it is planned to increase after re-construction of installation.

The experimental values of the ultimate solid coolant continuous flow velocity, the heat-transfer factor as a function of the heating wall temperature and core neutronics analyses point out the feasibility to build the reactors with reactor core cooling by the solid coolant that is based on the carbon with thermal-power density up to 15 MW/m^3.

In principle, the reactor consists of the reactor core, the coolant hopper above the core, the heat exchangers allocated lower than the reactor core and the elevators/lifts that off-take the cooled solid coolant from below and drive it to the upper hopper. Thus, the coolant moves throughout the reactor core and heat exchangers under the gravity and returns back to the upper hopper by the elevator. The pressure of helium in the reactor core is a little bit lower that the atmospheric one.

Some reactor core components, the schemes of flow velocity regulation, the possibilities of reactor core emptying in case of shutdown of all lifting mechanisms as well as the adjustment of control rod drivers and reloading systems might be implemented in different manner and are the subject for future research and optimization.

Of course, while considering the design versions of reactor with the solid coolant the principal deficiencies are taken into account, namely;

- Lack of natural convection;
- Difference in dynamic and static density, possible variation of dynamic density in streamline of immovable components of reactor core and heat-exchanger;
- Deterioration and, as a consequence, the possibility of arising split or broken particles.

At the same time, the reactors with solid coolant might meet rather wide range of "User Requirements", in particular:

- Reduced probability of reactor component damage due to lack of overpressure in the reactor pressure vessel, low flow velocity of coolant and lack of considerable internal stress;
- Minor consequences of potential emergencies because of inherent safety of the Reactor Facility as well as sufficient sub-critical margin.
- Sufficient capabilities for studying and full-scope simulation of all operational and emergency regimes are available for such reactors;
- Reactor Facilities have a small metal capacity, good neutron balance, potential of free allocation of starting materials for obtaining new fissionable isotopes;
- No need to use the rare and expensive materials in the Reactor technology, such as chrome, nickel and zirconium;
- High heat economy additionally reduces the cost-per-unit and waste heat to the atmosphere;
- Use of high burn-up particle fuel, the possibility to use thorium due to availability of free space in the reactor core and nearby gives reason to consider this Project to be advanced from the viewpoint of nuclear non-proliferation.

7. INNOVATIVE PROJECTS WILL MAKE SAFETY MORE RELIABLE

The example of solid coolant potential use examination gives hope that another projects can be suggested as well. It is assumed that they would be of considerable difference with the existing one and, therefore, create a capability for long-term stable development of nuclear power technologies.

The matter concerns, first of all, the development of fuel elements based on another conceptual background as well as producing other concepts for the fast breeders. All the same, the stage of experimental proof is obligatory.

The proposed Guidelines INPRO Document, essentially, does not contain any attempts to establish more strong requirements with regard to Nuclear Installation Safety rather than the best from the existing one. This work should be done later on basing on proposals available. In this meaning the given example of solid coolant use reveals a precedent for making the safety regulations tougher. Such guidelines tightening is a logical consequence of two reasons:

- Desire of human society to live in more safe world;
- Increase the scale of the nuclear power industry. The growing up of its use shall not rise the real risk for all groups of population all over the world.

No matter how extensively nuclear power is used, its risk to the public should be infinitesimal. Effective innovative projects should help increase the scale of the nuclear power industry.

8. CONCLUSIONS

In our view the Final Draft Guidelines for the Evaluation of Innovative Nuclear Reactor and Fuel Cycles developed under the aegis of IAEA is an important and timely effort. User Requirements are formulated so that the nuclear power development problems, its role and capabilities are clear to the general public.

Regulatory agencies, being a part of user community, are responsible for forming of mandatory regulatory requirements for devices and processes.

The current stage of innovative technology selection requires extensive interaction between the regulators and engineers, designers, environmentalists, investors and politicians in order to formulate the basis for regulatory requirements imposed on innovative technologies and calls for international collaboration among various licensing authorities in order to formulate "Licensing Infrastructure" of their licensing activity and develop mutually acceptable underlying principles, approaches and, if possible, unified standards.

Representatives of developing countries where nuclear power will be developing at a faster than average pace will have to play an important role in this endeavor.

New regulatory standards should ensure that safety enhancements precede the increase in a scale of nuclear power development. It would be reasonable to develop such standards separately for individual innovative technologies on a long-term basis, for instance, for 15 years.

Releases and discharges into the environment especially releases of long-lived Alpha-sources require careful study. Given an increase in the scale of nuclear power utilization, such discharges may have a significant cumulative effect.

Preliminary analysis shows that the new technology - the use of solid coolant in nuclear reactors - could meet broad range of User Requirements for innovative technologies and, thus, serve as the basis for sustained development of nuclear power industry on an increasing scale.

It seems that development of innovative technologies requires a concerted effort of the world community.

INNOVATIVE REACTOR TECHNOLOGIES - ENABLING SUCCESS

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Abstract. Many innovative reactors are being discussed, offering advantages in economics, sustainability, environmental impact, versatility and efficiency. To be successful, however, innovative reactors must meet the requirements for a successful build project. This requires achieving the mixture of innovation and proveness required to meet the first-of-a-kind hurdle. Based on the successful CANDU 6 reactor, a design still being built today, the ACR adds specific innovations in key areas chosen to achieve a balanced design. Capital cost has been significantly reduced by optimising the reactor-core design and simplifying systems. Key changes in this area include a move from a heavy water coolant to a light water coolant, and the adoption of SEU fuel. Construction times have also been reduced by using a modular design that takes advantage of modern construction techniques. Operating performance has been enhanced through improvements in system materials, equipment layout and component specifications. In parallel with these priorities, design adaptations have been applied so as to increase safety margins and defence-in-depth, again adding to the confidence in ACR licensability. The ACR development plan includes early review by regulators to reduce licensing risk, with international regulatory review having commenced. Overall, this places the ACR in a good position to meet the first-of-a-kind challenge, a necessary condition to enabling the success of an innovative reactor. AECL sees a logical evolution from the ACR, via increasing temperature and pressure capability, to the SCWR (Supercritical Water Reactor). AECL's CANDU-X program is already looking at designs for this concept. Inherent features of both ACR and the fuel channel SCWR lend themselves to different fuel cycles for the future. One of the prominent characteristics of the heavy-water moderated fuel channel reactor approach is the high potential for innovation. The evolutionary path allows innovation in practical steps, yet allows far reaching improvements to be ultimately achieved.

1. INTRODUCTION

Many innovative reactors are being discussed in forums such as the present IAEA symposium, targeting advantages in economics, sustainability, environmental impact, versatility and efficiency. These innovative approaches are being developed with a view to long-term sustainability through linkages to fuel cycle development. To be successful, innovative reactors must combine short-term and long-term benefits together. In the short-term, innovative reactors must meet the requirements for a successful build project. This requires achieving the mixture of innovation and proveness required to meet the first-of-a-kind bundle, so that initial build projects can proceed.

AECL has been pursuing innovation in reactor design using an evolutionary approach. This enables manageable, short-term innovative steps, but retains a long-term direction that can extend the evolution of design in far-reaching ways. The Advanced CANDU Reactor, or ACR design is the logical next step in the CANDU fuel-channel reactor design process, and achieves major improvements in economics while expanding safety margins. The ACR design is evolved from AECL's current CANDU 6 design with a specific series of enabling technologies which have been developed at AECL and are now being proof-tested.

The technology applied in the ACR is also the start of the series of steps that represent the long-term development path for CANDU fuel channel reactors. The ACR, as a light-water cooled fuel channel reactor, shares much in common with other light water reactors. In the same way the further evolution of the CANDU line fits in with the long-term development directions identified in initiatives such as the Generation IV International Forum (GIF) and INPRO. Specifically, AECL sees the product evolution from the ACR, via increasing pressure and temperature, to the Supercritical Water-costed Reactor (SCWR), one of the concepts identified by GIF for further development. AECL's "CANDU-X" program, looking 25 years into the future, has identified the heavy-water-moderated fuel channel SCWR as the natural evolution of the CANDU family. This would have a number of natural advantages as a readily developed, economical SCWR with a high level of safety assurance.

Inherent features of both ACR and the fuel-channel SCWR lend themselves to the adaptation to different fuel cycles of the future. Efficient use of neutrons, on-power refuelling, and simple, adaptable fuel bundle designs all enable fuel cycle benefits in the future.

One of the prominent characteristics of the heavy-water moderated fuel channel reactor approach is the high potential for innovation. The evolutionary path allows innovation in manageable steps, yet enables far-reaching improvements to be ultimately achieved.

2. BACKGROUND TO DEVELOPMENT STRATEGY

AECL has focused on the CANDU fuel channel reactor design, and its development potential over many decades, and continues to see this as a centrepiece of nuclear development. As part of the pressurized water reactor family, CANDU's share many characteristics with other light water reactors, while retaining a set of distinctive features:

- High pressure water coolant in individual fuel channels, with low-pressure, low temperature moderator
- Horizontal fuel channel design with on-power refuelling
- Simple, easily-fabricated fuel bundle design
- Use of heavy water to improve neutron efficiency.

The original impetus for these features was to enable a practical nuclear energy system to be developed in Canada in the decades following the Second World War, when the available industrial infrastructure was limited. The result was a nuclear reactor design which eliminated the need for a large high-pressure reactor vessel, which enabled all reactor core instrumentation and control elements to be located in the low pressure, low temperature moderator, and which enabled very low uranium requirements (and, with heavy water coolant, eliminated the need for uranium enrichment).

The subsequent decades leading up to today have seen many changes: in the global industrial infrastructure; in the understanding of uranium and other fuel resources; in safety and environmental protection expectations. With these changes, the basic principles underlying the CANDU approach remain valid, while design adaptations can readily align the CANDU system with current and future economic realities.

Since the original CANDU reactors were first built, the advent of deregulated, competitive energy markets has strongly emphasized the importance of low capital cost and short construction time. At the same time, increased availability of uranium resources and enrichment technology has decreased the pressure on fuel costs, while strategic, security-of-

supply considerations stress the need for fuel adaptability during the lifetime of the next generation of reactors.

The ACR design innovations are chosen to respond to this evolution of energy markets, while retaining the proven features of the CANDU line.

3. DESCRIPTION OF ACR PRODUCT

The ACR-700 design is an evolutionary development of familiar CANDU technology, adding a carefully chosen set of innovations to the major improvements in economics, operations and safety margins. With a gross electrical output of approximately 728 MWe, the ACR follows the same size range as AECL's standard CANDU 6 design, allowing much of the extensive experience base in CANDU 6 design, construction and operation to be utilized.

The ACR-700 design is rooted in the proven principles and characteristics of the CANDU system, and uses standard features of CANDU pressure tube technology built up over many decades of operation:

- Modular horizontal fuel channel,
- Available simple, economical fuel bundle design,
- Separate cool, low-pressure heavy water moderator with back-up heat sink capability,
- On-line/at power fuelling,
- Fuel cycle flexibility with high neutron efficiency,
- Passive moderator/shield tank heat sinks surrounding the pressure tube core,
- Two robust, quick acting, passive shutdown systems.

The following key features, derived from the enabling technologies, are incorporated into the design concept of the ACR

- Slightly enriched uranium fuel (nominally 2% U-235), contained in CANFLEX bundles to achieve burn-up (approximately 20 MWd/kgHM) and with further increases as operational experience increases,
- Light water replacing heavy water as the reactor coolant,
- More compact core design with reduced lattice pitch, reducing heavy water inventory and providing a highly stable core neutron flux,
- Enhanced safety margins, due to optimized power profile and void reactivity,
- Higher coolant system and steam supply pressure and temperature resulting in an improved overall turbine cycle efficiency,
- Reduced emissions, due to radiolysis of heavy water,
- Improved performance through advanced operational and maintenance information systems, and improvements to project engineering, manufacturing and construction technologies.

A simple diagram of the ACR-700 design is shown in Figure 1, and main design parameters are given in Table I.

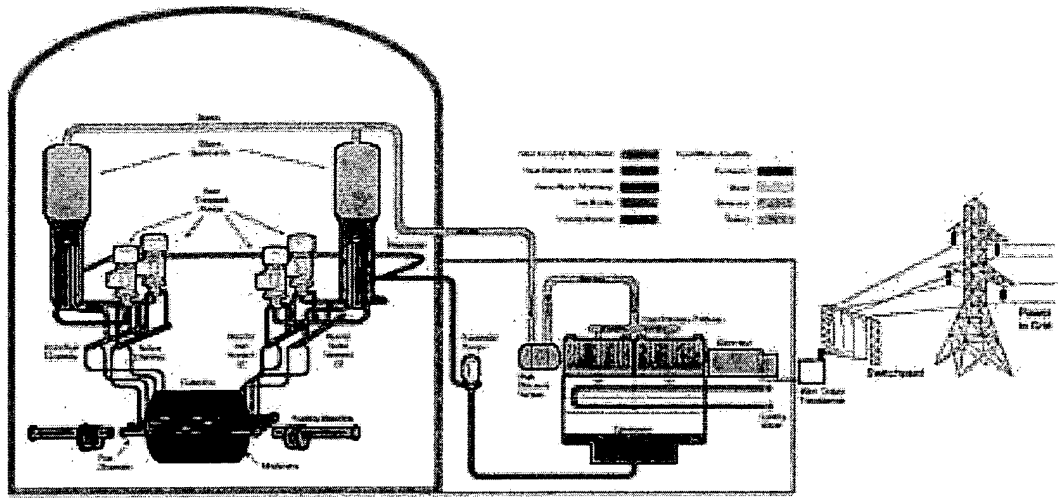

FIG. 1. CADDS diagram of ACR coolant system.

Table I. ACR main design parameters

Reactor thermal output:	1983 MW (th)
Nominal plant electrical output:	728 Mwe (gross)
Reactor coolant system pressure	12MPa (at reactor outlet header)
Reactor coolant system inlet temperature:	278.5° C
Reactor coolant system outlet temperature:	325° C
Nominal boiler steam pressure:	6.5MPa (a)
Nominal boiler feedwater temperature:	218° C
Number of fuel channels:	284
Number of fuel bundles per fuel channel:	12
Fuel design:	43-element CANFLEX fuels

The ACR has been designed from the initial conceptual stage with both constructability in mind.

The ACR is designed for highly modular construction. The entire reactor building internal equipment is assembled as a series of 105 modules, to be installed in "open-top" approach using a very heavy lift crane. The building arrangement allows the longest lead-time equipment to be installed at the latest date. This approach, coupled with the comprehensive use of a suite of electronic engineering and project tools, focussed on 3-D CADDS plant models, allows rapid construction and overall project schedules.

With a construction period (first concrete to start of main commissioning) of 36 months, the total project schedule for a first-of-a-kind ACR is 60 months, from contract effective date to in-service. For replica units (so called "n'th units") this is targeted to reduce to 48 months.

The ACR is also designed for ease of operation. The improved operation information systems, simplified system design, and layout for easy access, mean that the operating cost is reduced. By taking advantage of on-power refuelling, the plant is designed for a three-year interval between outages, and a target planned outage duration of 21 days.

4. DESCRIPTION OF FUEL CHANNEL SCWR

4.1. CANDU X

Research underway on the advanced CANDU studies new, innovative, reactor concepts with the aim of significant cost reduction and resource sustainability through improved thermodynamic efficiency and plant simplification. The so-called CANDU-X concept retains the key elements of the current CANDU designs, including heavy-water moderator that provides a passive heat sink and horizontal pressure tubes. Improvement in thermodynamic efficiency is sought via substantial increases in both pressure and temperature of the reactor coolant. Following on from the new Next Generation (NG) CANDU, which is ready for markets in 2005 and beyond, the reactor coolant is chosen to be light water but at supercritical operating conditions. Two different temperature regimes are being studied, Mark 1 and Mark 2, based respectively on the continued use of zirconium or on stainless-steel-based fuel cladding. Three distinct cycle options have been proposed for Mark 1: the High Pressure Steam Generator (HPSS) cycle, the Dual cycle, and the Direct cycle. For Mark 2, the focus is on an extremely simple direct cycle.

Supercritical water (SCW) as reactor coolant becomes feasible with development of the high-efficiency thermallyinsulated fuel channel. The use of compact core lattices allows replacement of heavy water coolant by light water, and becomes viable using a new bore seal for channel closure. By reducing the diameter of end fittings, the channel-to-channel lattice pitch can be adjusted to achieve zero or negative coolant void reactivity, and also fine-tuned with slight shifts in fuel enrichment.

We believe the pressure tube concept allows for great flexibility in the design of an SCWR, as the density, power and flux profiles can be optimized using the standard CANDU interlaced flow paths. Moreover, the use of SCW is not new, since some existing coal plants already use SC turbines at power ratings in excess of 800 MW(e) and attain overall cycle thermal efficiencies of > 40%. Supercritical coolant pressures permit large changes in enthalpy with small changes in temperature without encountering the two-phase region with its critical-heat-flux limitations.

The current CANDU X concepts are divided into two groups:

- Mark 1: uses zirconium based fuel cladding for high neutron-efficiency. Avoidance of excessive corrosion places a limit on coolant core-outlet temperature of some 420O C. Higher temperatures might be possible if the cladding surface is successfully treated with a thin corrosion-resistant layer.
- Mark 2: trades some neutron efficiency for thermodynamic efficiency: stainless-steel-clad fuel permits coolant outlet temperatures as high as 625O C, consistent with attaining the greatest efficiencies at the highest inlet temperatures of modern SC turbines.

To help guide the development of CANDU technology for deployment in the 2025-2030 timeframe, AECL maintains its CANDU X program. The CANDU X is an advanced concept targeted for deployment in the 2025 timeframe. The CANDU X is heavy-water moderated CANDU that is cooled by supercritical water (SCW). The use of SCW is not a new concept in power generation, having been used in fossil-fired plants for over 30 years. It is new, however, for nuclear plants, where a number of development challenges have to be overcome. Fortunately, CANDU reactors are very amenable to using SCW compared to LWRs for two reasons:

- Because the coolant and moderator in a CANDU are separate, the reactor is relatively insensitive to the large changes in coolant density that can occur across the core of a SCW cooled reactor.
- Designing a reactor core to handle the increased pressures and temperatures associated with SCW cooling is much simpler with a fuel-channel reactor than a pressure-vessel reactor.

In addition to these technical advantages, some potential safety advantages evolve from a move to SCW in a CANDU-type reactor.

5. FUEL CYCLE OPTIONS

At present, the primary nuclear fuel cycle worldwide is a once-through uranium cycle. At today's uranium and enrichment prices, the front-end and operating costs of this fuel cycle represent a small fraction (typically less than 10%) of total energy cost, and are therefore very competitive. Based on a once-through cycle, substantial energy content remains in the used uranium fuel. However, competitiveness of recycle options, led by MOX, remains uncertain, based on high forecast prices for reprocessing and continued low uranium and enrichment prices forecast into the future.

This means that the next generation of innovative plants will start their life with once-through uranium as the most likely fuel cycle of choice. However, given plant lifetimes of 60 years, it is likely that fuel cycle economics will change during the life of these plants. Therefore the ability to adapt to different fuel options is an important design attribute to consider.

In keeping with the CANDU tradition of neutron efficiency, the ACR is highly adaptable to a range of fuel cycles. The ACR is a very effective user of MOX fuel. The ACR can burn a 100% MOX core, without change to permanent reactor equipment, and can transition from a conventional SEU core to a MOX core without shutdown, using on-power refuelling. As ACR fuel burnups increase, the cost of MOX fuelling decreases. MOX fuel is relatively straightforward to manufacture in ACR fuel bundles.

Further, the ACR is a practical user of thorium-based fuels. A number of viable options for thorium fuelling have been identified, using the advantages of on-power fuelling and simple bundle design.

Thorium requires a core that includes fissile driver fuel. The use of on-power refuelling enables such a mixed core to be readily adjusted, and allows thorium fuel bundles to be shuffled so that they can be used as is, without the need for reprocessing to extract U-233 material.

The fuel channel SCWR core design will retain these attributes, since it continues with the common CANDU-based features. In addition, by increasing thermal efficiency, a further benefit in electricity product per unit of burnup is obtained.

6. FUEL CHANNEL REACTOR DEVELOPMENT POTENTIAL

Cooperative international activities such as GIF and INPRO are exploring long-term nuclear development options and identifying approaches to set development priorities. Typically, development review activities look at desirable end-points for global systems or for technologies. AECL has been part of this activity in support of Canada's role in the global nuclear community. As an organization responsible to establish a business case for reactor development, AECL has also reviewed the practicality of the development path – an important consideration in prioritizing development strategies. In this regard, the heavy-water moderated fuel channel design approach offers significant advantages, as follows:

6.1. Economical development

The development program for the ACR and onward to the fuel channel SCWR is a relatively economical activity:

- The evolutionary approach means that for each new design, a limited set of new technologies or components need be demonstrated. All the features of the SCWR do not need to be developed at once; rather, the ACR will form the originating technology framework, supported by development of materials for use at steadily increasing temperatures until the SCWR final conditions are reached.
- The modular fuel channel design approach means that the basic technology unit to be developed -- the fuel channel itself -- is relatively simple and small in scale. Testing of a single fuel channel represents a full core in scope. Similarly, the small, simple fuel bundle design can be readily developed and test irradiated at low cost.
- As part of the LWR family, ACR and SCWR development builds on the extensive existing investment in water reactor technology in Canada and around the world.

6.2. Economical deployment

The characteristics of fuel channel reactors lend themselves to economical build of prototype and first-of-a-kind units. Tooling and manufacturing development costs for modular components such as fuel channels are more readily absorbed in production runs of multiple components.

Further, by staging development in a small series of manageable, evolutionary steps, each successive design stage can use a large amount of design, licensing, equipment and construction technology from the previous stage, minimizing first costs. The ACR-700,

developed as an evolutionary step from AECL's current CANDU 6 design, keeps first-of-a-kind costs low enough to allow economically competitive deployment right from the first unit.

6.3. Low development risk

The evolutionary development approach reduces risk as well as cost. By moving toward the SCWR in a series of steps, the innovations in each step represent a modest technology risk which can be assessed through testing and demonstration and then performance in first units.

The fuel channel approach also has inherent risk reduction advantages. Individual fuel channels have been, and will continue to be designed for replacement if necessary; other core components in the low-pressure, low temperature moderator environment, see modest duty which is easily replicated in tests, and are also readily designed for adjustment or replacement. Finally, fuel designs, in particular alternate fuels, are easily demonstrated in a power reactor environment due to on-power refuelling and the simple CANDU fuel design. A single channel, or set of demonstration channels, can be use to demonstrate successive adaptations in fuel design or increases in burnup.

6.4. Scope of development potential

Relative to other current reactor types, the heavy water moderator fuel-channel reactor family has an extremely broad scope for development. For most of the past, development has been focused on refining and improving the familiar natural-uranium fuelled CANDU line. While extensive development of enriched uranium and other alternate CANDU fuels has been completed, ACR represents the first optimization of the CANDU concept to take advantage of slightly enriched fuel. The result is a significant opportunity for economic benefits, broadening the safety envelope and system simplification. Once the ACR is established, further improvements can be introduced, such as increased thermodynamic efficiency via increased steam pressures and reduction in fuelling overall costs via increased burnup.

6.5. Flexibility

The fuel channel concept has characteristics naturally suited to fuel cycle flexibility, as noted above. Innovative designs are tailored to ensure this flexibility is maintained or extended. In addition, there is great flexibility in deployment. Manufacturing of CANDU fuel and plant components can be readily introduced to host countries, because of the simpler, smaller-scale nature of the key components.

Finally, the modular nature of the fuel channel approach means that the same stage of technology can be deployed in different power levels. For example, AECL has established the ACR technology in both a reference ACR-700 (700MWe) design and an alternate ACR-1000 (1000MWe) design, each with the same system and component technology and the same licensing basis. The technology can be readily adapted to larger or smaller power outputs, depending on market demand.

6.6. Synergies with other technologies

Looking into the long-term, and at an evolving future, more than one innovative nuclear technology is likely to be needed. In particular, in order to balance the demands of economical energy with long-term conservation of fuel resources, a blend of "breeder" and "burner" technologies would be needed. As a fuel-efficient burner technology, heavy-water moderated fuel channel reactors represent a good complement to breeder technologies as they are

developed. As part of the water-reactor design continuum, fuel channel reactors can make use of, and contribute to, a very wide R and D and experience base.

7. CONCLUSIONS

AECL has established a development path forward for the CANDU family of fuel channel reactors. The next step, the Advanced CANDU Reactor (ACR) design, offers significant improvements in economics and safety case, and retains the traditional CANDU flexibility in adapting to alternate fuels. For the longer term, the fuel channel reactor type is an attractive option for supercritical water reactor (SCWR) applications. Development of fuel channel reactors along this path offers some important benefits in: economical development and deployment; low development risk; scope of development potential; flexibility; and in synergies with other technologies.

STATE OF THE ART OF SECOND PHASE FEASIBILITY STUDY ON COMMERCIALIZED FAST REACTOR CYCLE SYSTEM

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Abstract. Japan is preceding the development of the fast reactor (FR) and related fuel cycle system as a priority target to secure its own energy supply. In addition, the commercialization of FR cycle system can be expected to contribute greatly to security of energy resource and preservation of global environment in the world. To deliver this attractive feature, the feasibility study on commercialized fast reactor cycle system (Feasibility Study) was launched in 1999 by the organization composed of Japan Nuclear Fuel Cycle Development Institute (JNC), Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI). The Feasibility Study aims to present the prominent FR cycle technologies for commercialization by around 2015. In consideration of future social needs, five development targets (ensuring safety, economic competitiveness, reduction in environmental burden, efficient utilization of resources and enhancement of nuclear non-proliferation) were set up at the start point. The first phase Feasibility Study was carried out in the period from 1999 to 2000 and the second phase launched in April 2001. The period of the second phase study is for five years. Feasible candidates screened out in the first phase study are being investigated with the adoption of innovative technologies to clarify promising candidates. In this paper, the evaluation method for the clarification of the promising candidates is briefly discussed, and the progress of the design study and experimental tests for key technologies is presented.

1. INTRODUCTION

In the 20th century, civilization developed drastically by rapid evolution of science and technology. Although peoples of advanced countries have enjoyed materially rich life, the fears of energy resource depletion and environmental destruction have been gradually actualized. In addition, it is expected that the total population in the world amounts to about 10 billion until the middle of the 21st century. As for the energy demand, it is possible along with this increase in population to increase by about 1.5 to 2 times more than at present.

Supposing this energy need is boarded with fossil fuels such as oil and natural gas, it worries not only the shortage of fossil resources, but also the influence on global warming caused by the exhaust of carbon dioxide. These have adverse impacts on the living conditions. Moreover, focusing on developing countries, the growth of large energy demand is expected and the global scale aggravation to energy and environmental problems has been worried in the 21st century.

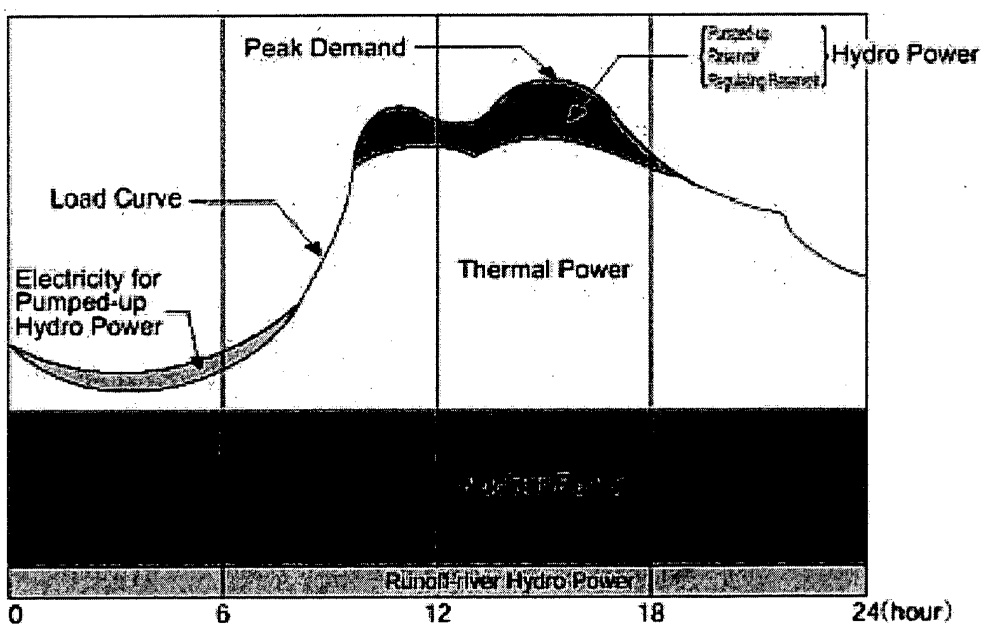
Since Japan is very poor in energy resources, it is necessary to save them to supply with stability for a long period. Furthermore, the technology development with small environmental burden is indispensable in energy production. To deal with the problems on energy supply and demand in the 21st century, Japan is promoting the improvement of energy efficiency and the development of renewable energy technologies.

As for nuclear energy, the deployments of new reactor plants are stiffened due to the anxiety to safety and the uncertainty of radioactive waste disposal. On the other hand, the necessity

of nuclear power is reviewed under the backgrounds of the recent steady operation records of nuclear plants, the necessity of energy supply with the global environmental preservation, and the progress of the disposal business, etc. In the international corporations of GEN-IV and INPRO, the research and development of nuclear power system including fuel cycle for next generation are in progress.

Nuclear power has no greenhouse gas emission and the amount of waste for a unit energy generation is extremely little. The spent fuel recycle by the introduction of reprocessing is one of the promising candidates for sustainable energy sources of the 21st century. Especially, the effective use of uranium resource is spectacularly improved by the fast reactor (FR) cycle. The energy supply by nuclear power becomes possible for a long term of hundreds of years or more, therefore the FR cycle is expected as a future nuclear power system.

As Japan is very poor in natural resources and imports 80% of them, it is very week in domestic energy supply and is not good to depend on the only energy source. Electric power companies in Japan seek the optimal combination of power sources as shown in Figure 1. Power companies currently use nuclear power as their base load supply, because it is better for economic performance and supply stability. Especially, the cost of oil and natural gas is relatively higher compare to other countries, thus the price competitiveness of nuclear power is higher in Japan. The optimum mixture of energy supply is important so that the government supports the development of various kinds of energy sources, such as hydro power, oil, natural gas, nuclear power, window power, photovoltaic power, and so on.



Source : Agency of Natural Resources and Energy

FIG. 1. Optimal combination of power sources in Japan.

In addition to economic performance and supply stability, nuclear power plants emit much lower carbon dioxide than that of solar and wind power, as well as thermal power of fossil energy sources, such as oil, natural gas and coal. The Kyoto Protocol on curbing global warming (COP3) determined the legal binding reduction target to advanced countries and it showed to reduce about 5% of greenhouse effect gases in all advanced countries. In order to stabilize the amount of greenhouse gas emission, increase of nuclear power plant and introduction of non-fossil energy and countermeasure of energy saving are necessary. There are currently 51 nuclear power plants in operation in Japan, producing 34% of electricity. As for the renewable non-fossil energy, Japan is a leading country in the world of photovoltaic power generation and wind power generation capacity has been rapidly increasing for the last several years.

2. FEASIBILITY STUDY ON COMMERCIALIZED FR CYCLE SYSTEM

The FR cycle is important as the technology that achieves long-term use for nuclear power aiming at resource saving and little waste by recycling spent fuels. In Japan, realization of the FR and related fuel cycle system is a priority target to secure its own energy supply. In addition, the commercialization of FR cycle system can be expected to contribute greatly to security of energy resource and preservation of global environment in the world. To deliver this attractive feature, the Feasibility Study (FS) on commercialized fast reactor cycle system is being performed by the organization composed of Japan Nuclear Fuel Cycle Development Institute (JNC), Japanese electric utilities, Central Research Institute of Electric Power Industry (CRIEPI) and Japan Atomic Energy Research Institute (JAERI).

Figure 2 depicts the development steps of FS for FR cycle systems. FS aims to present the prominent FR cycle technologies for commercialization by around 2015. The first phase FS was carried out in the period from 1999 to 2000. A concept of innovative recycle system that can efficiently reprocess and fabricate TRUs, and burn them in the FR is studied. As a results of the first phase in about two years, several promising FR plants and the related fuel cycle systems have emerged as candidates for the future FR cycle system which can attain the development targets for commercialization.

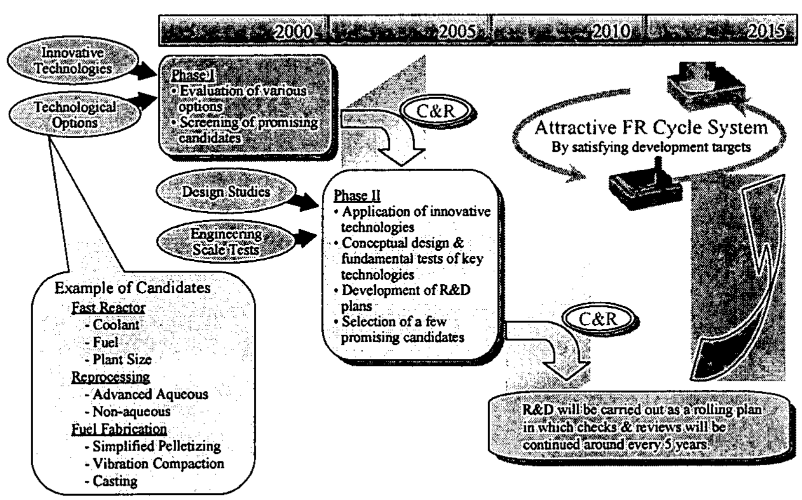


FIG. 2. Development steps of feasibility study for FR cycle system.

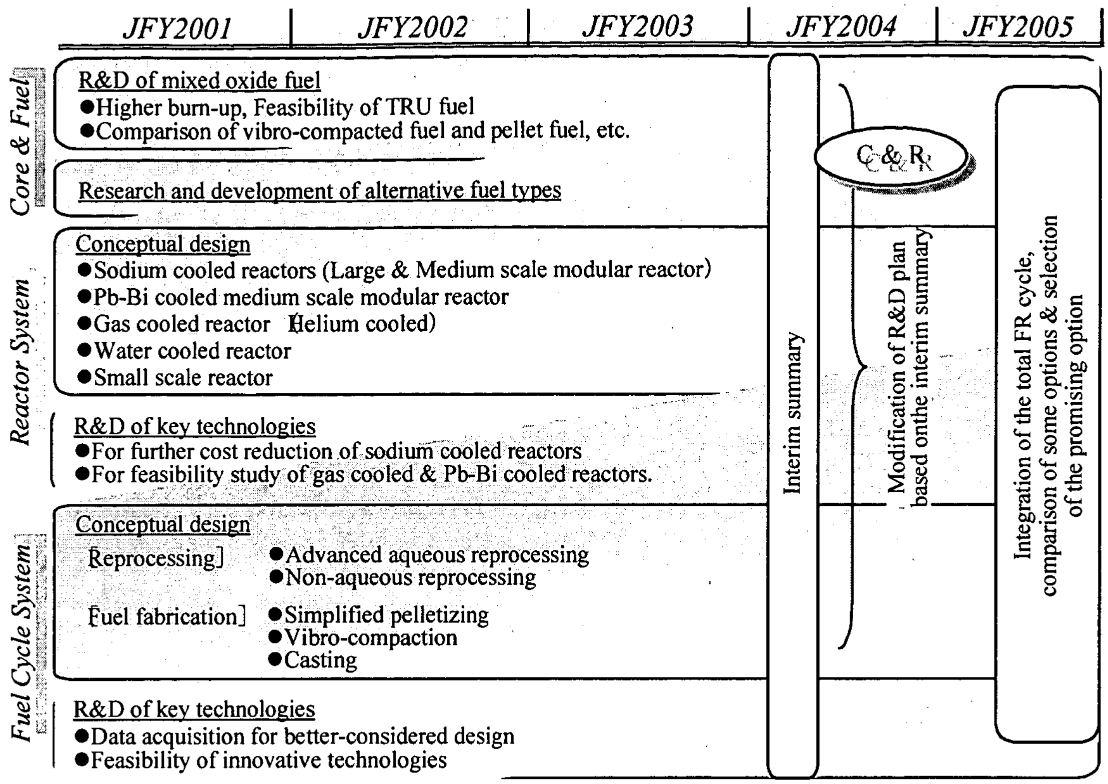


FIG. 3. Scope of the second phase feasibility study.

Following the first phase FS, the second phase launched in April, 2001 as shown in Figure 3. The period of the second phase FS is for five years. Feasible candidates screened out in the first phase are being investigated with the adoption of innovative technologies [1-7]. In the second phase FS, the large variety of design study and experimental tests of key technologies for each candidate are being conducted to confirm the feasibility of candidates. The interim report of the second phase FS will be drawn up in JFY 2003 and the framework of promising candidates and related roadmaps will be indicated in consideration of the technical consistency between FR and fuel cycle system.

International cooperation is important and effective as well as domestic collaboration in developing the FR cycle system. FS is being conducted in cooperation with the related research organization in Europe, U.S.A. and Russia as shown in Figure 4. The information exchange concerning the design and fuel development for helium gas-cooled FR has been being well achieved between CEA and JNC. As for lead-bismuth-cooled FR, the fundamental experiments of corrosion/erosion characteristics against structure are underway by a collaborative work with FZK/KALLA. The EAGLE project, which is aiming at demonstrating the effectiveness of inner duct structure to enhance the fuel discharge without propagating neighbor subassemblies and to obtain a perspective for re-criticality free concept in sodium-cooled MOX fuel core, is in progress between .NNC (National Nuclear Center)/RK (Republic of Kazakhstan) and JNC with Utilities. In addition, Japan is nominated as leading country for the development of GEN-IV SFR (Sodium-Cooled Fast Reactor System) [8].

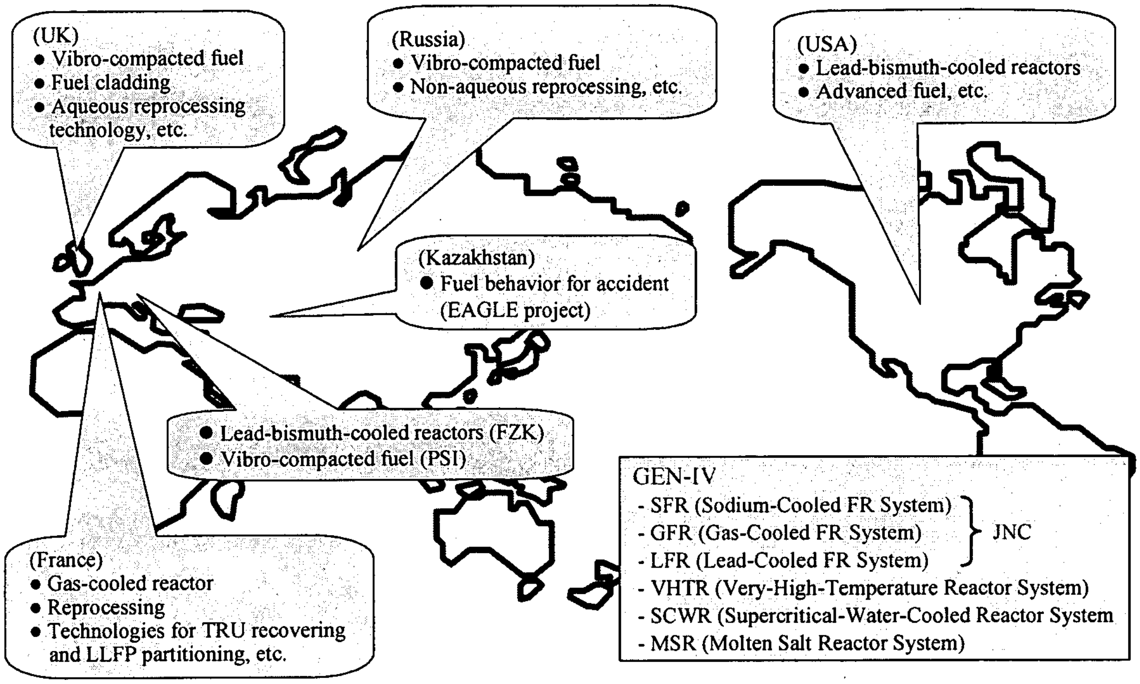


FIG. 4. International cooperation.

In this paper, the progress of the design study and experimental tests for key technologies will be presented, and the evaluation procedure for the promising candidates will be briefly discussed.

3. TARGETS FOR FEASIBILITY STUDY

On the major premise of safety, it is expected to the energy system of the 21st century to be able to reduce the environmental burden and to supply energy for the long term. The improvement of economy is indispensable to be built upon business under electric power trade liberalization. In addition, nuclear proliferation resistance is requested from the international aspect to prevent nuclear weapon diffusion. Under such circumstances, the FR cycle technology excels in neutron economy, high energy supply ability, excellent TRU (transuranium) burning and LLFP (long-lived fission product) transmuting characteristics etc., and has potential that satisfies the above-mentioned requirements.

In consideration of future social needs, five development targets (ensuring safety, economic competitiveness, reduction in environmental burden, efficient utilization of resources and enhancement of nuclear non-proliferation) were set up at the start point of FS. Needless to say, ensuring safety is the basic premise in the development of FR cycle system. Economic competitiveness, as today's most important issue, has to be drastically improved by the development of innovative technologies for the implementation of FR cycle system. The radioactive waste generation, as one of the crucial issues of fission energy use, has to be improved in the process for the handling of long-lived radiotoxicity. The contents of the five development targets are described below and summarized in Figure 5.

- **Safety**
  - Risks caused by introduction of FR < Risks that already exist in society
- **Economic Competitiveness**
  - Power generation cost ≦ Future LWRs, other energy resources
  - To ensure cost competitiveness in the global market
- **Reduction of Environmental Burden**
  - To reduce impact of high level radioactive waste of repository by means of burning or transmuting long-life nuclides (TRUs and LLFPs)
  - To reduce radioactive waste generated in the course of plant operation and maintenance as well as decommissioning
- **Efficient Utilization of Natural Resources**
  - To produce sustainable nuclear fuel through applying the advantage of the neutron economy
  - To respond to diverse needs for energy resources (Production of hydrogen, desalination of sea water, heat supply, dispersion of power sources, etc.)
- **Enhancement of Nuclear Non-proliferation**
  - To reduce burden of nuclear physical protection and safeguards (Isolate pure plutonium in any FR cycle process)
  - To effectively operate non-proliferation system (Remote monitoring system, etc.)

FIG. G. Development targets of commercialized FR cycle systems.

3.1. Ensuring safety

Safety is a major premise for not only nuclear systems but also any engineering systems to be accepted in the society. In the development and operation of FR cycle system, it is essential to examine how to ensure the safety of every facility in its design, construction, operation and decommissioning stages, keeping in mind potential risk due to the existence of a great deal of nuclear fuels and radioactive materials inside.

We adopt the safety design that has basic philosophy of defense-in-depth and gives top priority to the prevention of occurrence and expansion of abnormalities. With this design framework, we keep the safety at higher level than or equivalent to that of the light water reactor of the same generation.

Furthermore, based on the characteristics of each facility, we aim at building more certain and clearer safety measures. For this purpose, a passive safety mechanism is installed or strengthened to prevent core disruptions. In the event of a hypothetical core disruption, re-criticality should be avoided and the event should be terminated inside the reactor vessel or the containment vessel. Moreover, the safety design takes into consideration physical and chemical characteristics (chemical activity, radiotoxicity etc.) of materials handled in a nuclear reactor plant or a fuel cycle facility.

The safety design aims at keeping the risk of introducing FR cycle system small enough comparing to the risks already existing in the society.

3.2. Economic competitiveness

For commercial use of the FR cycle system, it is essential to achieve the economy that allows the introduction of the system based on the principle of market mechanism. For this reason, the target on this issue is to have a competitive edge in power generation cost over future competing energy sources. Furthermore, aiming at world-class cost competitiveness, we look to overseas procurement for the further improvement of economics.

3.3. Reduction of environmental burden

In the commercial use of the FR cycle system, it is necessary to maximize its attractiveness, such as excellent thermal efficiency, the small amount of waste per energy generation, and the reduction of greenhouse gas emission etc. Taking advantage of favorable neutron economy of the FR cycle system, there is a possibility to reduce exposed dose and risk by the geological disposal of radioactive wastes. Thus, we investigate the separation and transmutation of long-lived radioactive elements (TRU, LLFP) accumulating in fuel cycles to reduce the radioactivity and the potential toxicity of the high level waste. It is also addressed to reduce the volume of radioactive wastes generating in operation, maintenance and decommission of cycle facilities.

3.4. Efficient utilization of resources

The long-term energy demand in the world is predicted to increase. However, there are various uncertainties in energy supply and demand predictions. FR has the excellent features of efficient burning of TRU including higher order plutonium isotope as well as favorable neutron economy. These features enable us to use nuclear power as sustainable energy source over the long period of hundreds years or more by recycling uranium resource. The establishment of the FR cycle technology as one of the energy options means to be able to correspond to these uncertainties flexibly. In addition to use FR as base load power supply, we investigate various FR business chances, such as distributed power supply, heat supply, hydrogen production etc.

3.5. Enhancement of non-proliferation

In the commercial use of the FR cycle system, we have to openly and clearly show the peaceful use of nuclear energy to the global society, which eliminates the risk of nuclear materials being diverted to nuclear weapons. In order to secure peaceful uses of nuclear materials, nuclear facilities in Japan accept full inspections by the government and safeguards of IAEA based on Non-Proliferation Treaty (NPT). In addition to these extrinsic barriers, the highly radioactive material contaminated fuel provides proliferation resistance feature as an intrinsic barrier. Thus, we investigate a reprocessing system with no pure plutonium in all processes. The remote maintenance, surveillance and fuel fabrication technologies are also important to handle minor actinides and fission products contaminated fuels. The enhancement of intrinsic barrier expects to lead to reducing the correspondence to safeguard.

4. COMPREHENSIVE EVALUATION METHOD

4.1. Flow of comprehensive evaluation

Figure 6 shows the flow chart of comprehensive evaluation in FS for various FR cycle concepts. Development targets and design requirements were set up at the starting point, and then design studies and key technology tests are being conducted to meet those targets and requirements. R&D results are summarized in the technical database system for FR cycle to evaluate the achievement levels to the design requirements from technical viewpoint.

Design information on the core, reactor, reprocessing and fuel fabrication systems (which are the units of the design work) is combined to create attractive and/or representative FR cycle concepts. Taking into consideration their rationality as a combination, similarity of cycle characteristics and compatibility between the FR system and the fuel cycle system, candidates of FR cycle concepts are clarified as shown in Figure 6 [9].

The promising candidates of FR cycle system are comprehensively evaluated from viewpoints of compatibility with development targets, technical feasibility and social acceptance. Followed by the evaluation, key FR cycle concepts are clarified, which is developed towards commercial introduction, and then the R&D plan for future FR cycle systems is submitted.

4.2. Evaluation indices

A set of system evaluation criteria corresponding to FS development targets is being studied to provide measures to objectively judge the performance of FR cycle system concepts. Table II depicts seven evaluation indices prepared in FS. In developing the evaluation indices, quantification was considered wherever possible and the evaluation indices were stratified using a multi-layer hierarchy to clearly indicate the relationship between the development targets and the indices.

4.3. Qualitative evaluation method

Considering the characteristics of evaluation indices and design information level available at present, the above seven evaluation indices are classified into three categories. The first category is those indices that can be quantified, including "Economics", "Resource utilization" and "Environmental impact". The second category is the indices that are prerequisite for nuclear use, and it is essential to meet the evaluation criteria depending on the system characteristics. "Safety" and "Proliferation resistance" are included in this category. The third category is the indices that are hard to be quantified and needed the judgment based on the expert judgment. This category includes "Technical feasibility" and "Social acceptance".

For the indices in the first category, hierarchical structure is developed as shown in Figure 7. The utility function U(X), ranging from 0 to 1 of its value, is allocated to each attribute in the lowest layer, and estimates non-dimension worth. Weighting factors are set to attributes so that the total weighting factor in one level should be one, to estimate the attribute in the upper level. The weighting factors are set based on engineering judgment. With the same procedure, we estimate non-dimension worth of the higher-level attribute in the hierarchy. The final non-dimension value means the achievement level to the development target corresponding to this evaluation index.

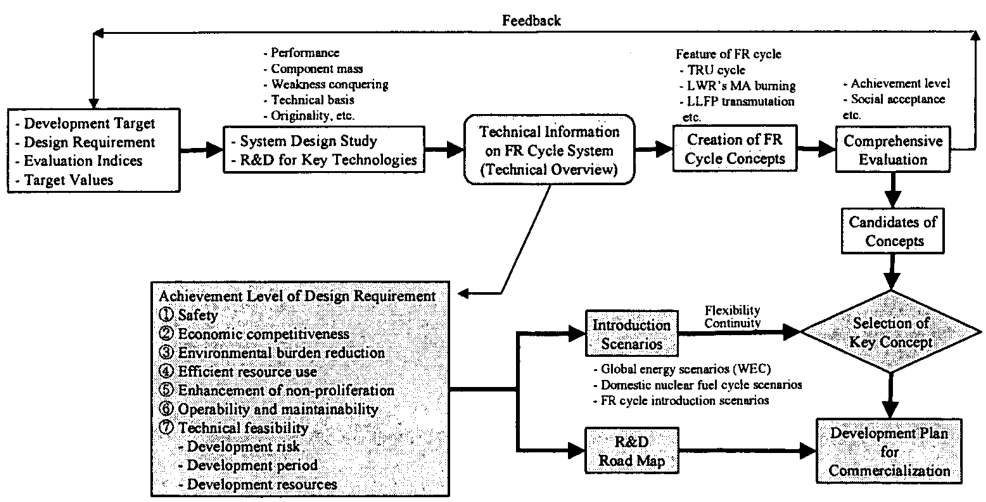


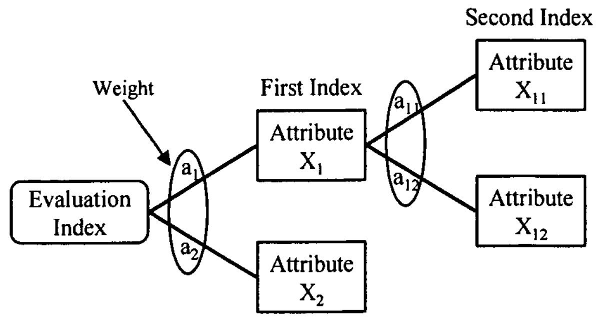
FIG. 6. Schematic flow of comprehensive evaluation.

Table. II. Evaluation indices for the feasibility study.

Development targets	Evaluation indices	First index	Second index	Third index
Economic competitiveness	Economics	Power generation cost (yen/kWh)	- Investment cost - Operation and maintenance cost - Fuel cycle cost, etc.	
Efficient utilization of resources	Resource utilization	Cumulative uranium demand (tonU)	- Out-pile cycle time - TRU inventory, etc.	
		Efficiency of uranium use (%)	- Burn-up - Recover factor, etc.	
Reduction in environmental burden	Environmental impact	Concentrate and retain	- Waste volume	- High level waste - TRU waste - Low level waste, etc.
			- Waste volume converted repository area	
			- Radioactive toxicity	
			- Exposure dose (Repository site)	
		Dilute and disperse	- Exposure dose (Surrounding)	- Ocean - Sea
Ensuring safety	Safety	Design basis events (DBE)	- Guide line for safety design - Safety assessment	
		Margin for beyond DBE	- Prevention of core damage - Elimination of re-criticality - In-vessel retention (PAHR)	
Enhancement of non-proliferation	Proliferation resistance	Intrinsic factors	- Isotopic composition of Pu - Diversion barrier of process, etc.	
		Extrinsic factors	- Safe guard - Physical protection, etc.	
-	Technical feasibility	Development risk	- Technology level	
			- Difficulty level	
		R&D investment	- R&D budget	
			- R&D period	
-	Social acceptance	Benefit	- Responsiveness to social needs - Benefits to local community, etc.	
		Risk	- Scientific risks - Psychological factors, etc.	

The utility function is defined through three points. The allowable performance has a non-dimension value of zero corresponding to the lowest limit for introducing the FR cycle. The sufficient performance of commitment level has a value of 0.5 to smooth introduction of the FR cycle. The challenging target has a highest value of one to promotion or ideal level. We adopted an exponential function with three constants A, B and C as shown in Figure 8 [10]. Figure 9 shows a case of power generation cost, where these three points are specified as follows:

— Allowable limit: 10 yens/kWh corresponding to the renewable wind power
— Commitment level: 4 yens/kWh corresponding to the future FR cycle
— Challenging target: 3 yens/kWh corresponding to the present natural gas thermal power in overseas



$$U = a_1U_1 + a_2U_2(X_2)$$
$$= a_1\{a_{11}U_{11}(X_{11}) + a_{12}U_{12}(X_{12})\} + a_2U_2(X_2)$$

U : Utility Function on the Index

FIG. 7. Hierarchical structure of evaluation index.

As for the "Safety" in the second category, we confirm that each technology meets the judgment standards specified by "Guide line for safety design", "Safety assessment" and "Margin for beyond design basis events (prevention of core damaged, elimination of re-criticality, in-vessel retention and post accident heart removal) ". These standards should be a premise to be the subject of FS. As for the "Proliferation resistance", we prepare a checklist that indicates the evidence of engineering judgment by experts. About the "Technical feasibility" in the third category, we estimate "Development risk (technology level and difficulty level)" and "R&D investment (R&D budget and period)" based on the expert judgment. For the "Social acceptance", we examine the significance and the necessity of the FR cycle, such as externality of environment effect, comparative evaluation with other power sources, social risks, return-on-investment evaluation and introduction scenarios of the FR cycle. We also plan to examine the benefits and risks for individual and community in consideration of psychological factors.

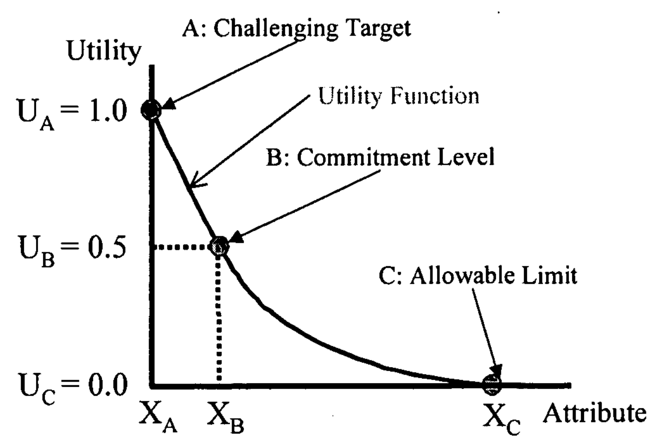


FIG. 8. Utility function.

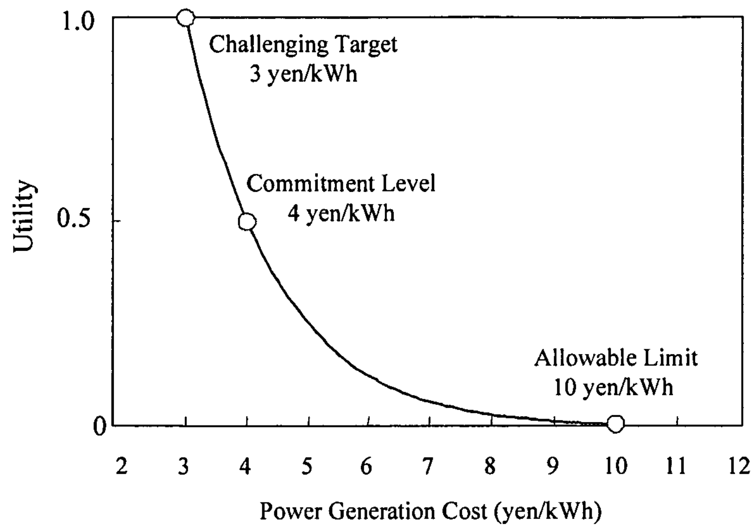


FIG. 9. Utility function of power generation cost.

5. CURRENT STATUS AND FUTURE WORK

5.1. Fast reactor systems

5.1.1. Sodium-cooled fast reactor [11-29]

With regard to system study, the conceptual design of advanced loop-type sodium-cooled FR was confirmed focusing on the improvement of economic competitiveness (Fig. 10). In order to achieve the reduction of the construction cost, the plant design has been investigated from three approaches, such as scale merit, standardization and learning effects, and the design improvement by employing the innovative technologies.

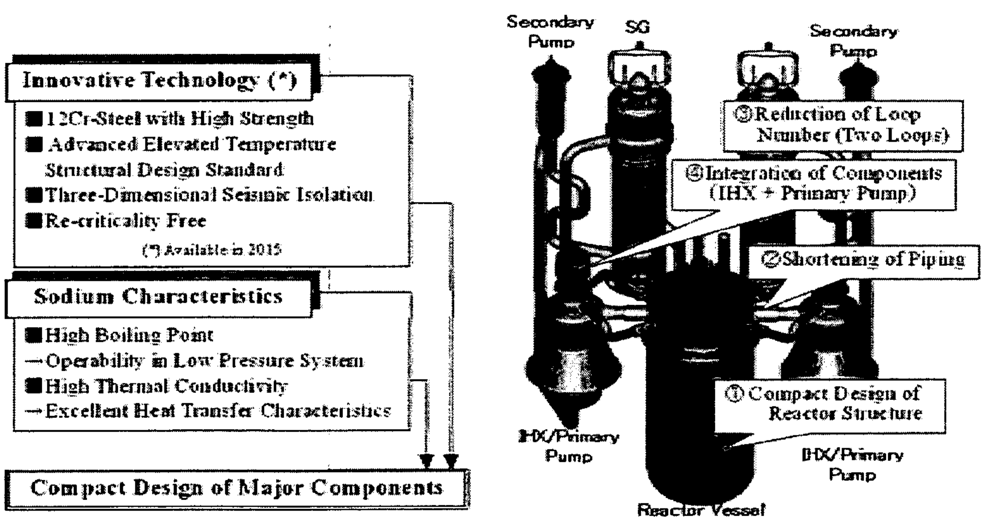


FIG. 10. Design improvements of sodium-cooled fast reactor.

R&D activities for key technologies related to the advanced loop-type reactor concept are: compact reactor vessel (R/V), large diameter piping under high flow velocity condition, and integrated components with a primary pump and an IHX. The compact design of R/V results in high sodium velocity in the upper plenum. Scaled model water tests are being conducted to seek the design measures for stabilizing the coolant flow in the plenum and preventing gas entrainment at the free surface. The two-loops design raised several R&Ds related to thermal hydraulics, such as flow induced vibration issues. These difficulties are facilitated by higher flow velocity in one loop. A scaled model water test of hot leg piping is preparing to obtain the prospect of solution. The integrated components may enhance the fretting wear of IHX tubes. The evaluation methods for the fretting wear in 12Cr steel and related experimental data are necessary to be investigated to ensure its integrity. A scaled model test is preparing to evaluate the vibration of IHX tubes induced by pump.

5.1.2. Helium gas-cooled fast reactor [30]

A carbon-dioxide gas-cooled FR and a helium gas-cooled FR were evaluated and compared with each other. For a gas-cooled FR, high temperature up to 850°C with direct cycle system has a great potential to accomplish the development targets especially in economic competitiveness. Meanwhile the combination of the carbon-dioxide gas-cooled FR and gas turbine power generation would be very difficult to commercialize due to the problem of structural material corrosion. As the results, the helium gas-cooled FR with coated-particle fuel was selected because of its superiority of high plant efficiency.

Features of Helium-cooled Fast Reactor

① High-temperature/high thermal efficiency:	Multi-purpose utilization (Efficient power generation, Production of hydrogen etc.)
② Safety:	Potential for securing high degree of safety with passive safety features
③ Improvement of economic competitiveness:	Deletion of secondary cooling system and turbine buildings by adopting direct cycle system and common use of equipment by adoption of twin plant

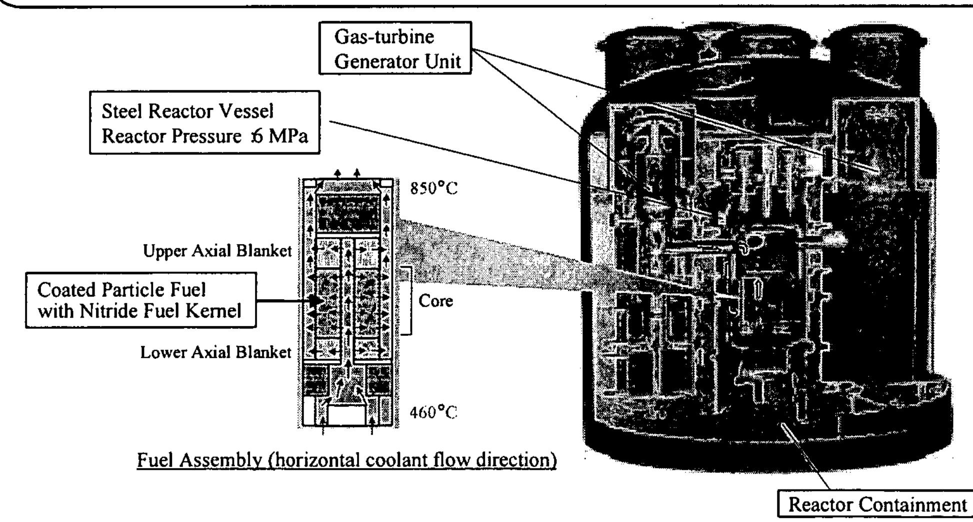


FIG. 11. Helium gas-cooled fast reactors.

With regard to the coated-particle fuel, a commercialized fuel concept for thermal neutron high-temperature gas reactors exists that uses spherical particles made by multi-layer coating an oxide fuel with high-density graphite layers or SiC to give the fuel particles high heat capacity and hard coating layer. However, as previous test data indicate that the coating layers will be damaged with a burn-up of less than 10,000 MWd/t in fast neutron environment. Thus TiN was selected as a candidate of most probable coating material in consideration of thermal, mechanical and irradiation characteristics against fast neutrons.

The most valuable feature of high temperature resisting coated-particle fuel is the high possibility of implementing countermeasures against reactor core disruption. Core melt and re-criticality should be avoided without any active component actuations even under depressurized accident conditions. Figure 11 shows the schematic of helium gas-cooled FR. The design study is in progress focusing on the achievement of a core melt proof reactor safety concept.

5.1.3. Lead-bismuth cooled fast reactor [31-40]

The heavy metal-cooled reactor is a relatively new plant concept that has only quite recently begun to be studied even in other advanced countries with the exception of its application in nuclear submarines in Russia. Therefore, FS evaluated heavy metal-cooled reactor concepts using data from previous studies conducted in Russia. At first, we selected a tank-type reactor because a loop-type reactor would require technically difficult measures to alleviate thermal stress for the piping due to high specific gravity. Next, we selected a medium-scale reactor because it would be difficult to achieve the economic competitiveness with a large-scale reactor due to the very large weight of core support structure and the difficulty of meeting the mandated standards to withstand earthquake.

With regard to the coolant, lead-bismuth was adopted because a lead-cooled reactor would require maintenance at high temperature of about 400°C due to its high melting point. To compare the effect of coolant driving system on capital cost, the 55 MWe reactor of natural circulation type and the 75 MWe reactor of forced convection type were examined. The forced convection reactor system has slight economical advantage because the mass of unit power is smaller than that of the natural circulation type. Figure 12 depicts the forced convection type lead-bismuth-cooled FR.

For the lead-bismuth-cooled FR, the prevention of material corrosion in high temperature coolant is the most crucial issue of judging the technical feasibility. Therefore, tests have started to seek measures to control the corrosion of core and structural materials in collaboration with FZK and domestic universities. In the collaboration studies with FZK under stagnant lead-bismuth condition, ODS and 12Cr-steel showed good compatibility with lead-bismuth eutectic under a proper oxygen content of 10^{-6} mass% at 550°C and bellow. However, the thickness of the oxide layer was getting thinner with temperature increase over 600°C. Beyond 570°C, dissolution attack was observed at some portions. It is estimated that the oxide layer becomes to loose its adhesion to the material. The application temperature range of the existing materials will be confirmed by corrosion tests. As the results obtained up to now, the development of a new anti-corrosive material is indispensable for the commercial use of the lead-bismuth-cooled FR.

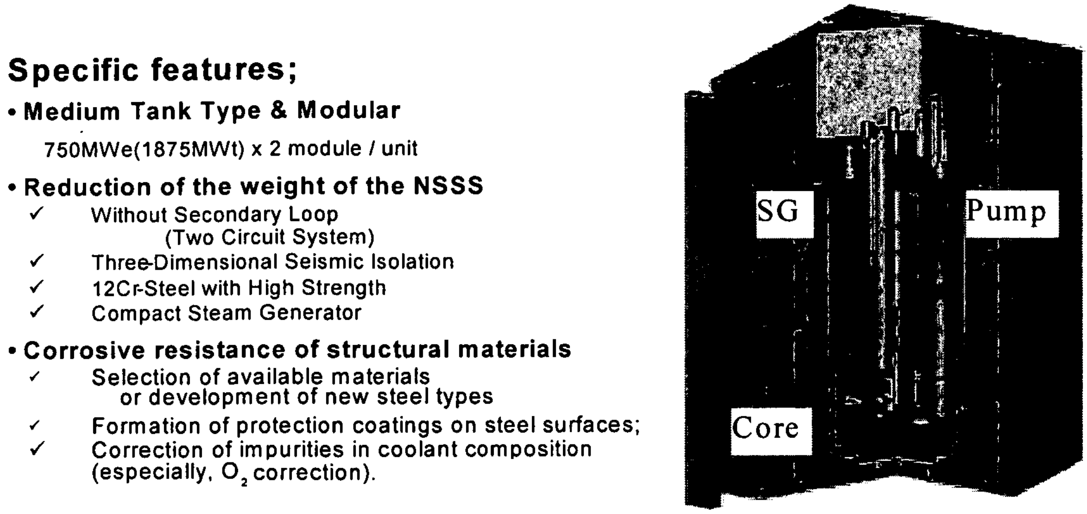


FIG. 12. Lead-bismuth-cooled fast reactor (medium-scale/tank-type/modular/forced convection.

The use of nitride fuel is necessary for the lead-bismuth-cooled FR to achieve high burn-up and breeding performance. With regard to nitride fuel, the important tasks to be addressed are: the collection of basic data through fuel irradiation tests, the economic concentration and collection of ^{15}N, and the analysis of high-temperature characteristics including the transitional characteristics such as the mechanism of nitrogen dissociation.

Since there are many issues to be solved for the lead-bismuth-cooled FR, it is considered that the long-term R&D works are needed until commercial use.

1.1.1. Water-cooled fast reactor [41-42]

The water-cooled fast reactor was designed by referring to a high conversion ratio BWR (Boiling Water Reactor) and increasing the density and void fraction of the core to increase the fuel volume ratio. In terms of economical competitiveness, it is considered that this concept will be equivalent or slightly superior to ABWR (Advanced BWR) because of the elimination of the recirculation pumps, the reduction in the containment vessel volume, etc.

The design base safety analysis and evaluation confirmed the possibility of attaining required safety level. However, it is necessary to clarify the core cooling performance during an accident and to evaluate the risk of re-criticality in the event of a core disruption, including the sequence of events and the behavior of the molten fuel during core disruption, and the performance of the containment function during re-criticality.

In terms of core fuel, Pu breeding ratio of about 1.04 can be achieved with low decontamination and high Pu enrichment fuel (about 35%). The multi-layer compound core

system is adopted by using the fuel pin with inner blanket in the central core section. The Pu inventory of the compound system is about two times large compared to that of a sodium-cooled fast reactor.

It is necessary to confirm the thermal hydraulic and mechanical characteristics of dense core layouts as well as the safety performance during core disruption and to evaluate the possibility of further improvement in the burning capability of low-decontamination TRU fuel.

5.2. Fuel cycles

The current status of R&D for fuel cycle systems is shown in Figure 13. In order to accomplish the development targets, the fuel cycle system needs to be composed of simplified and rationalized processes. Furthermore, dealing MA (minor actinide) and LLFP as well as U and Pu in the closed cycle system, it is indispensable to adopt the innovative technologies to establish the fabrication process of low-decontamination factor TRU-bearing fuel, and to develop the remote operation and maintenance system in a hot cell facility. Main technologies for reprocessing have been broken down into categories such as U-recovery, U/Pu/Np-recovery and MA-recovery, and for fuel fabrication into categories such as fuel fabrication and stacking/compaction as shown in Figure 14 [43-46].

Reprocessing

Advanced Aqueous Method (Oxide, Nitride Fuel)	● Establishment of advanced aqueous method by hot tests for crystallization technology, TRU recovery technology, etc. ● Identification of alternative technologies by hot tests for supercritical fluid direct extraction method , etc.
Oxide Electro-winning Method (Oxide Fuel)	● Confirmation of technological feasibility by small-scale Pu tests
Metal Electro-refining Method (Metal, Nitride, Oxide Fuel)	

Fuel Fabrication

Simplified Pelletizing Method (Oxide, Nitride Fuel)	● Development of simplified pelletizing technology and ascertaining of productivity of low decontamination TRU fuel by trial tests
Vibration Compaction Method (Oxide, Nitride Fuel)	● Development of gelling method, rolling granulation method, etc. and selection of promising granule production method ● Development of vibration compaction technology by small scale system tests using granule made by gelling method, electro-winning method, etc.
Casting Method (Metal Fuel)	● Development of casting technology and ascertaining of the possibility of the feasibility by domestic small-scale tests

FIG. 13. Current status of R&D for fuel cycle systems.

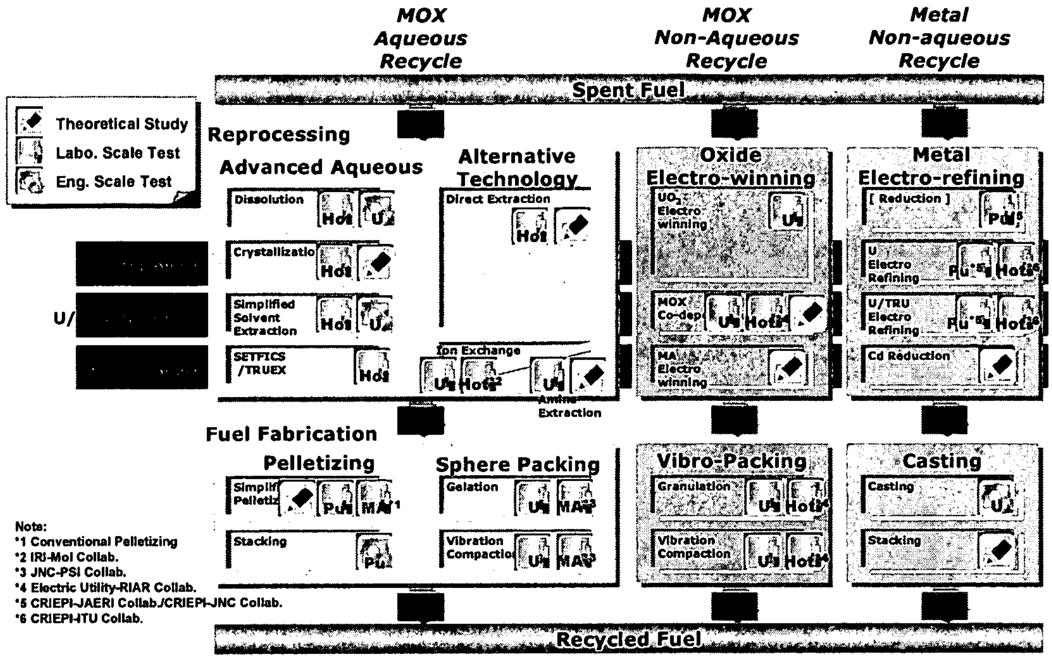


FIG. 14. Status of recycle technology.

5.2.1. Advanced aqueous process [47-55]

The advanced aqueous process consists of a simplified PUREX process with the addition of a uranium crystallization step and the SETFICS process as MA recovery process. The features of the process are the following:

- The purification process of U and Pu in the conventional process is eliminated, and
- U/Pu is co-extracted with Np with reasonable decontamination factors (DF) for recycle use,
- The uranium crystallization removes most of the bulk heavy metal at the head end and eliminates it from down stream processing,
- The main process stream is salt-free, which reduces the secondary waste,
- Neither separated Pu nor radiation-free nuclear material exists in any step of the entire fuel cycle.

In the simplified PUREX process, Np recovery in mixed U and Pu product solution has been demonstrated with DFs over 1000 in small-scale hot tests in Chemical Processing Facility (CPF). It should be studied to optimize U/Pu/Np recovery condition and the DF for fission products. In the crystallization process, the dissolved solution is cooled down and excess U is precipitated as a crystal of uranyl nitrate hexahydrate (UNH) according to the solubility at the low temperature. It is expected that the decontamination factors for fission products in the UNH product are approximately 100 from a simulated dissolver solution test. The combination of SETFICS/TRUEX process using TBP and CMPO is applied to the system as the MA recovery process. Small-scale hot tests were implemented to investigate the

separation efficiency of MA from lanthanides. It was confirmed with cold tests that a salt-free reagent such as hydroxylamine nitrate (HAN) was applicable in this process.

Besides the above described process, some alternative techniques have been also investigated; supercritical fluid direct extraction method as the alternative for the dissolution, U recovery and U/TRU recovery, the amine extraction method as the alternative for the SETFICS, and the extraction chromatography method for the SETFICS.

5.2.2. Pyro-electrochemical process [56-58]

Candidates for advanced reprocessing are the modified pyro-electrochemical processes, which are based on Russian-RIAR and US-ANL methods. They offer several presumed advantages, the most important among them are the following:

- Ability to process refractory and hot fuels, due to the high solvating power of molten salts and the radiation-resistant features of the chemical reagents involved (no organic radio-sensitive molecules)
- Compactness (limited number of transformation steps – actinides is early recovered in the form suitable for recycling)

Some other features of the pyro-electrochemical process can be mentioned, like suitability for "Onsite processing".

5.2.2.1.Oxide electro-winning process

The reprocessing technology for oxide fuel is rather focused at the transition period from the current fuel cycle to the next advanced fuel cycle. The development of these key technologies focuses on the safe, reliable, industrial scale-up of electro-winning and refining systems including an extraction process for MA and LLFP.

Continuing efforts by RIAR have demonstrated a successful operation of the oxide electro-winning by using several kg of spent fuel from BOR-60. Japanese electric utilities and JNC are now trying to modify these processes. Especially, MOX co-deposition behavior has been demonstrated with highly-decontaminated UO_2 and PuO_2. However, it is known that some of fission products and cladding material disturb the MOX co-deposition behavior.

5.2.2.2.Metal electro-refining process

The R&D on metal electro-refining process has been principally designated to CRIEPI. Metal electro-refining is more effective for fast reactor metallic fuel cycle. Applying metal electro-refining process to the fast reactor oxide base fuel cycle, additional processes for the initial reduction of MOX and the final oxidation of metal are required. Fundamental studies are still required to adjust the electro-winning condition.

Development of long-life component materials, including crucible material, is an issue for realization of the dry process because of the corrosive and high temperature operating conditions. Safeguard ability of the dry process should be assured with the real time monitoring equipment and inspection system. The loss of fissile materials to waste should be minimized and the recovery of MA can be further optimized. The treatment of chloric type wastes has to be guaranteed for long-term stable storage. Concept optimization for industrial-scale spent fuel pyroprocessing is important to reveal weakness of existing pyroprocesses and clear the direction of the improvement.

JNC is arranging testing infrastructures in Tokai works for metal electro-refining as well as oxide electro-winning. Collaboration programs with domestic and overseas partners are in progress. The glove box equipment is being prepared by CRIEPI at CPF in JNC-Tokai Works to make integral Pu and U experiments related to metal electro-refining.

In small-scale process feasibility tests, which have been carried out by CRIEPI in collaboration with the Institute for Transuranium Elements (ITU) of the EU, the electro-refining tests for U-Pu-Zr ternary alloy fuel containing MA (non-irradiated) were performed and fundamental data, such as recovery ratio, were obtained. Also, tests for recovery of Pu by liquid Cd cathode were carried out using U/Pu ratio in salt as a parameter. It was confirmed that recovery of heavy metals at a concentration exceeding the design value (10wt%) is possible and the possibility of rationalization of the system was confirmed. Moreover, it should be studied to optimize U/Pu/MA recovery condition under fission product in liquid Cd cathode.

5.2.3. Fuel fabrication [59-60]

Fuel fabrication for the advanced cycle must be as simple as possible and suitable for the massive remote operation to handle radioactive materials, which are recovered from the reprocessing with low decontamination factors. Three candidates for fuel fabrication process are being investigated in FS; a simplified pellet process, vibro-packed process using particulate fuel and metal casting process. R&D for metal casing process is in progress by CRIEPI to modify the casting fuel fabrication method.

5.2.3.1.Simplified pellet process

MOX pellet fabrication technology based on glove box confinement has been verified in highly decontaminated plutonium recycling system in the commercial LWR and fast reactor fuel cycle. However, this process must be modified to fabricate low DF fuels in a remote operation mode. Simplified pellet process is the shortest route in which MOX powder adjusted Pu content is co-converted directly by microwave heating process from Pu, U and Np nitric acid solutions for the next pelletizing process with minimum pre-treatment of powder. The key in this technology is to prepare the well-homogenized and controlled powder to obtain a high throughput.

When the simplified pellet process is applied to low DF products, the pellet design specification should be relaxed to realize the operability of the fabrication system in a hot cell facility and to match an increase of impurity level. In addition, it is necessary that fabricability of MA-bearing pellet fuel is established under low DF condition. JNC is conducting basic parameter tests to optimize powder characteristics and fabricate MOX pellet with the preliminarily simplified process.

5.2.3.2.Sphere and vibro-packed fuel fabrication

Sphere and vibro-packed fuel fabrication method is a most important technology common to the particulate fuel products from both aqueous and dry reprocessing in advanced fuel cycle. The concept of vibro-packed fuel itself was introduced about 40 years ago. The key issue is the selection and optimization of particulate MOX-MA fuel fabrication methods. These are gel precipitation, dry granulation process in simplified PUREX and MOX co-precipitation in oxide electro-winning.

The controlled fabrication method of two or three size distribution of granular particles is important to achieve an equivalent smear density with MOX pellet pin of about 80%TD. The challengeable technology is to get the smaller size particle of higher Pu content MOX-MA fuel controlled less than 100μm in diameter by remote operation. The experiences in BNFL and RIAR for the vibro-packed fuel suggest that fuel pin quality assurance should be optimized for the continuous operation of manufacturing and inspection process. Laser scanning system to guarantee the quality of particle fuel and three-dimensional CT scan system to check the smear density distribution in a fuel pin are still under basic investigation. The prevention of fuel-cladding chemical interaction (FCCI) is another issue. Controlling oxygen potential under an irradiation condition is necessary to achieve high burn-up capability. Both the initial conditioning of oxygen-to-metal ratio of MOX particulate and the mixing of oxygen getter with fuel particle are candidates.

5.2.4. Fuel cycle options

Considering the above elementary technology results and the experience of conventional reprocessing development, three fuel cycle processing technology options have been studied.

5.2.4.1.Advanced aqueous + Simplified pelletizing or vibration compaction

The conceptual design study on the system with plant capacity of 200t/y has been confirmed to satisfy the development targets. Moreover, regarding a small-scale plant of about 50t/y, in order to improve economic competitiveness, the following tests are being conducted on a laboratory-scale using spent fuel:

- Technology for alternating key parts of the system
- Technology for fabrication of MA-containing MOX pellets by remote operation

R&D is necessary concerning further reduction of waste, simplification of the system, and recovery of LLFP elements.

5.2.4.2.Oxide electro-winning + Vibration compaction

The conceptual design study with plant capacity of 50t/y has been confirmed. The following R&D is essential in order to confirm the possibility of technical feasibility of the system:

- Confirmation of the technical feasibility of the key technology (MOX co-deposition, MA recovery, etc.)
- Technology for manufacturing of uranium-added fine metal particles that control the performance of vibration compacted fuel

R&D is necessary concerning salt waste disposal and recovery technology for LLFP elements.

5.2.4.3.Metal electro-refining + Casting

The conceptual design study with plant capacity of 50t/y has been confirmed to satisfy the development targets. In order to improve performance of the key technology of the system, the reaction speed and recovery rate tests are being conducted on a laboratory-scale using the fresh MOX fuel. R&D is necessary concerning salt waste disposal and recovery technology for LLFP elements.

6. CONCLUSION

FS has been in progress aiming at commercialization of promising FR cycle system for base road power supply in Japan. Reactor core and plant system designs with various kinds of fuels and coolants have been studied to clarify the potential performance and the technical feasibility of FR cycle concepts. Crucial R&D items have been found out for each FR cycle concept and several experimental and/or analytical works have been underway. The perspective for the promising candidate concepts will be clarified preliminary at the end of March 2004, based on the on-going design studies and R&D results. The comprehensive evaluation method is expected to offer transparent and objective evidences to support the estimation. The significance on the international collaboration becomes greater than so far.

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NUCLEAR POWER DEVELOPMENT ON THE BASIS OF NEW NUCLEAR REACTOR AND FUEL CYCLE CONCEPTS

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Abstract. Much of the global demand for electricity and some demand for heat can be met by a nuclear technology that will comply with the safety, environmental and economic requirements of a large power industry. Nuclear power can grow on a large scale based primarily on big nuclear plants with fast reactors. The key requirements among those placed on the reactor and fuel cycle technologies include: efficient utilisation of accumulated Pu and reduction of specific U consumption by an order of magnitude or more; natural safety – deterministic exclusion of accidents involving large radioactive releases, balance between the radiation hazards of radioactive waste subject to burial and of uranium extracted from the earth; resistance to proliferation of nuclear weapons; reduction in the cost of new plants relative to modern LWRs. This presentation describes the work done on designing a plant with a demonstration lead-cooled 300 MWe reactor (BREST-OD-300) and on experimental validation of the adopted reactor and fuel cycle design.

1. INTRODUCTION

The "*Strategy of nuclear power development in Russia in the first half of the 21st century*" [1] maintains that the experience in using nuclear energy gained by mankind over 50 years provides a sound basis for developing and demonstrating early in this century a technology of fast reactors with a closed nuclear fuel cycle, that will meet today's requirements of large-scale energy production in terms of economic, environmental and safety characteristics, and will be capable of attaining the goals identified by nuclear scientists back in the 1940s.

Studies carried out by Russian scientists and engineers in the last 15 years [2] laid a groundwork for the Initiative "On energy supply for sustained development of mankind, radical solution of nuclear weapons proliferation problems, and global environmental improvement", which was voiced by President Putin at the UN Millennium Summit on September 6, 2000, as an invitation to international cooperation in these vitally important areas. With this aim in view, Minatom initiated an international project at the IAEA, referred to as INPRO.

The lessons drawn from the fifty-year experience and consideration of the new conditions suggest that the following main requirements should be placed on the NPP with a fast reactor and its closed fuel cycle:

— equilibrium fuel composition (BR=CBR≈1); provision of a Th blanket to produce U for thermal reactors at some later date, when cheap ^{238}U is exhausted;
— strong and convincing safety case for long-term and large-scale production of nuclear electricity, with accidents involving catastrophic radioactive releases deterministically excluded despite all possible human errors, equipment failures and external impacts;
— a strong and convincing safety case for burying radwaste for many thousands of years without upsetting the natural radiation balance;

- elimination of the weapons-grade material generation channels in today's nuclear power, such as radiochemical separation of Pu and, at a later point, isotope enrichment of U, - to go hand in hand with improvements in the international nonproliferation regime and physical protection measures;
- lower plant cost relative to the existing power units so as to make nuclear competitive with conventional energy sources.

Fast reactors with U-Pu fuel have a unique excess of neutrons, which is their fundamental physical resource for meeting the key requirements. Moreover, it allows finding an adequate technical concept (fuel, coolant, design, etc.) for the fast reactor, which will not be too different from the existing civil and military technologies but will be consistently based on the principles of natural safety, which means essentially reliance on the laws of nature.

To prove the feasibility of making an NPP consistent with the above requirements of a large-scale power industry, it is planned to build an experimental plant at the Beloyarsk NPP site, which will have a demonstration 300 MWe lead-cooled fast reactor (BREST-OD-300) with an on-site nuclear fuel cycle and a radwaste treatment complex.

The main working objectives of the BREST-OD-300 plant and its on-site fuel cycle include:

- exclusion of prompt criticality excursion by virtue of core physics and design (equilibrium fuel composition and reactivity margin of ~ß);
- demonstration of reactor resistance to severe accidents with and without operation of the control and protection system:
    - input of the whole reactivity margin;
    - trip of primary and secondary pumps;
    - rupture of steam generator tubes;
    - freezing - unfreezing;
    - coincidence of accidents;
    - extreme accidents;
    - all kinds of "made-up" accidents;
- launching of the on-site closed fuel cycle;
- operational refinement of the radwaste treatment processes.

In 2002, the BREST-OD-300 design was developed for the Beloyarsk NPP site, including [3]:

- engineering design of the BREST–OD–300 reactor facility:
    - reactor facility;
    - steam generator;
    - pump;
    - upper plate;
    - reactor vault;
    - reloading machine.
- engineering design of reactor facility systems for:
    - heating;
    - coolant intake, conditioning and injection;
    - pressure compensation;
    - radioactive gas treatment;
    - coolant treatment with gas mixtures;
    - air cooling of the vault;
    - normal and emergency cooling;

- confinement of steam generator leaks;

— NPP design:
  - general layout;
  - process design features;
  - main building;
  - turbine hall and secondary circuit;
  - structural design features;
  - construction management plan;
  - preliminary safety case;
  - environmental qualification;
  - survey for construction.

Engineering design of fuel cycle equipment has been completed for:

— cutting of fuel assemblies;
— fuel regeneration;
— fabrication of fuel rods;
— fabrication of fuel assemblies.

Engineering design of radwaste treatment facilities is finished.

The general view of the NPP with its on-site fuel cycle and the design of the BREST-OD-300 reactor are given in Figures 1 and 2. The reactor characteristics are found in Table I.

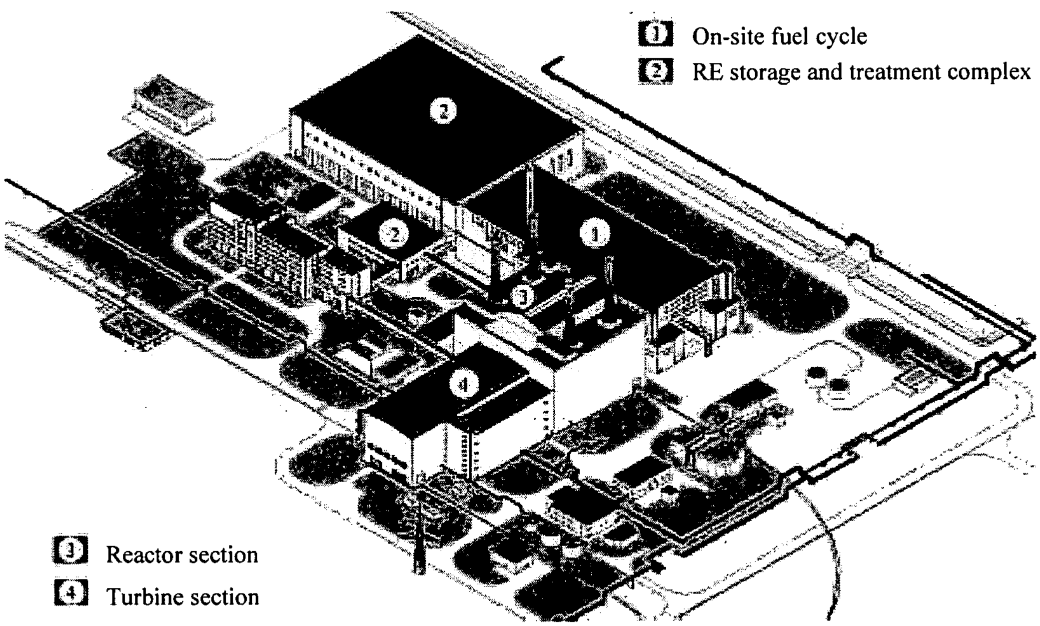
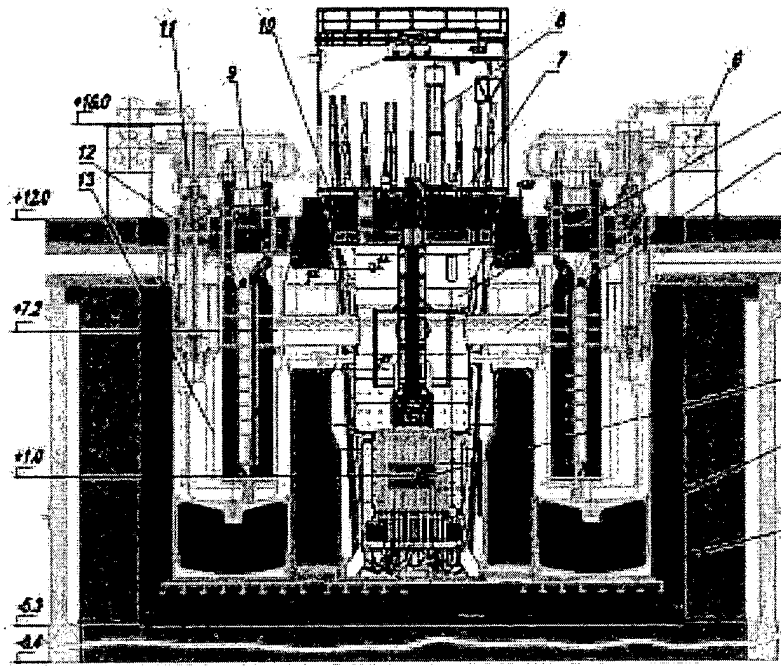


FIG. 1. General view of the NPP.



1 - core	*5 - core barrel*	*9 - steam generator*	*13 - filter*
2 - vessel	*6 - cooling system*	*10 - upper plate*	
3 - reactor vault	*7 - instrumentation column*	*11 - main circulation pump*	
4 - header	*8 - in-pile reloading machine*	*12 - SG-MCP unit*	

FIG. 2. BREST-OD-300 Reactor.

2. PRINCIPAL SAFETY FEATURES

To provide stability of BREST-OD-300 reactor in off-normal conditions, the core and the cooling circuits have been designed as follows [4]:

— fuel is mixed uranium-plutonium mononitride (UN + PuN) which features high density $\gamma \geq 13$ g/cm^3, heat conductivity $\lambda \sim 18$ W/(m·K), melting temperature T_{melt}=2800°C and phase transition temperature T_{phase}=1300°C;

— coolant is liquid lead which does not enter into exothermic reaction with water, air and structural materials, does not catch fire, is resistant to radiation, is low-activated, and allows heat removal at low pressure and with a large boiling margin (T_{boil}~2000°C at P~1 MPa);

— lead bond between fuel and cladding excludes their thermal-mechanical interaction, provides high heat conductivity of fuel rods and low operating temperature of fuel (T_{av}~620°C and T_{max}<900°C), and reduces release of fission gas and its pressure on the cladding as burning progresses;

— core composition and geometry and fuel properties provide for full reproduction of plutonium in the core (BR=CBR ~1), ensure small reactivity variation with fuel burnup ($\Delta\rho_{burn} \ll \beta_{eff}$), small power and total reactivity effects ($\Delta\rho_{tot} \sim \beta_{eff}$);

Table I. Technical characteristics of BREST-OD-300 and BREST-1200 reactors

Characteristic	BREST-OD-300	BREST-1200
Thermal power, MW	700	2800
Electric power, MW	300	1200
Core fuel	UN+PuN	UN+PuN
Fuel inventory, (U+Pu+MA)N, t	17.6	68
Fuel lifetime, eff. Days	1500	1500÷1800
Refuelling interval, eff. days	300	300
CBR	~1.05	~1.05
Average in/out Pb temperature, °C	420/540	420/540
Maximum coolant velocity, m/s	1.7	1.7
T_{clad}^{max} with allowance for overheating, °C	650	650
T_{fuel}^{max} with allowance for overheating, °C	870	880
Steam temperature at the steam generator outlet, C°	525	520
Number of steam generators/pumps	4/4	8/8
Superheated steam pressure at the SG outlet, MPa	25	24.5
Design service life, years	30	60

- a "sparse" square fuel lattice and shroudless fuel assemblies provide a large flow area for the coolant and high natural circulation, and help avoid loss of heat removal in the event of local flow blockage at the FA inlet;
- three-zone fuel profiling in the core – which is provided by the use of fuel rods with different diameters but the same fuel composition and rod pitch in FA (rods with a smaller diameter placed in the centre of the core and larger ones, at its periphery) - ensures uniform lead temperature gain and cladding temperature distribution across the core, allows stabilisation of these parameters and increases temperature margins;
- lead reflector, instead of uranium blanket, flattens power distribution, provides large negative reactivity effect in case of low lead level in the reactor, reduces density coefficient of reactivity, and rules out production of weapons-grade Pu;
- passive coolant flow feedback to reactivity is implemented by lead-filled channels, with levels therein depending on the lead head at the core inlet;
- passive threshold feedback to reactivity from coolant flow and temperature is implemented by hydraulic control elements which provide passive reactor shutdown in case of loss of circulation or high coolant temperature at the core outlet;

- lead circuit is designed to have a great heat storage capacity and high flow inertia, and is provided with a flow bypass to ensure natural circulation of lead in case of pump trips;
- passive removal of decay heat directly from the lead circuit by a system with natural air circulation for an unlimited time;
- configuration and parameters of the steam-water circuit prevent lead freezing;
- large volume of the gas plenum, steam discharge from the gas plenum to emergency condensers and on, to the stack, and a hydraulic valve installed on the reactor lid, prevent circuit overpressure and lid tear-off in a very unlikely event of multiple rupture of SG tubes.

The discussion below addresses accidents caused by initiating events that none of the existing reactors, including those under development, can survive (i.e. beyond-design-basis accidents), and accidents specific to BREST:

- input of maximum operating reactivity (TOP WS),
- total loss of forced circulation of lead (LOF WS),
- total loss of heat sink to the secondary circuit (LHS WS),
- overcooling of the primary circuit (OVC WS),
- coincident occurrence of initiating events (WS),
- loss of steam generator integrity,
- loss of integrity of the lead circuit and reactor building.

Several temperature levels indicative of the extent of fuel damage were chosen for accident analysis:

- fuel cladding temperature up to T_{clad}=800°C and fuel pellet temperature up to T_{pel}=1300°C - the core remains serviceable due to low stresses in fuel claddings;
- fuel cladding temperature T_{clad} ≤900 °C (for a short time, ~20s) – fuel rods remain intact; cladding temperature in the range from 900 to 1200 °C - fuel cladding failure, release of gaseous and volatile fission products into the circuit, damage to some structural elements of the reactor (an accident of so-called "economic" class, with loss of reactor core or reactor as a whole);
- T_{clad}>1200°C - damage to the reactor core, melting of fuel claddings (T_{melt}~1500°C), lead boiling (T_{boil}~2000°C), decomposition (T_{ph}>1600°C) and melting (T_{melt}~2800°C) of mononitride fuel, damage to many reactor structures and possible loss of primary circuit integrity with the ensuing release of radioactivity.

All these initiating events and reactor response to them are discussed in detail in Ref. [4]. Let us turn to the accident involving failure of external safety barriers.

One of the reasons behind the choice of heavy coolant rather than sodium for fast reactors is associated with accidents involving failure of external safety barriers (reactor lid, building), leading to long-term exposure of coolant to the atmosphere. Deterministic demonstration of the plant safety requires analysis of even such unlikely, but possible, events, proving that the extremely severe consequences of these accidents are ruled out.

No detailed analysis has been performed so far for such accidents at fast reactors. Despite the scarcity of the input data required for such analysis, rough calculations were performed to assess the consequences of the accident according to one of its possible scenarios for the 300 MW(e) fast reactors with sodium and lead coolants.

The essential differences in these two cases stem from the self-sustaining burning of Na when in contact with air. This event will cause substantial heat release equal to or exceeding residual heat release, high pressure of Na vapours (3 mm Hg at 500 °C as compared to 10^{-2} mm Hg with Pb), high radioactivity of sodium proper, and loss of fuel cooling as a result of Na burning.

As postulated, the accident involves reactor lid tear-off, destruction of the building, lead temperature increase to 700 ^{0}C, and sodium temperature rise to 500 ^{0}C. It is assumed that temperatures remain at these levels for four days, resulting in the maximum radioactivity release in case of lead coolant and minimum release in case of sodium, because lead temperature will decrease while sodium temperature will rise when heat removal can only rely on radiation and air convection. Calculations and experiments showed that when in contact with air, lead does not catch fire under T=1200 ^{0}C. Therefore, fuel rods are assumed to remain intact and radioactive release from fuel does not exceed the design limit.

It was assumed in both cases that chain reaction was suppressed, but the personnel failed to terminate radioactive release. The release from gas plenum in the beginning of the accident was disregarded in the analysis.

Lead entrainment was assessed from experimental data obtained in the studies on liquid lead evaporation and oxidation in the first 10 hours of lead-air contact when the rate of evaporation is at its maximum. In the accident under consideration, the rate of lead entrainment was 22 g/s. Under the worst weather conditions, lead concentration near the ground at a distance of 10 km from the site would be 0.02 mg/m^3, which is equivalent to 2 MPC (maximum permissible concentration) in terms of the chemical toxicity of lead (lead MPC = 0.01 mg/m^3).

Estimation of maximum release of hazardous radioactive nuclides in the course of the accident (during tens of days), made under the assumption that lead of Grade C00 will be used in BREST and that there will be no in-service purification of lead to remove these radionuclides, showed that total release of radioactivity would amount to ~2300 Ci. It should be noted that potential radiation hazard of the daily release of polonium - a mere 3 Ci - is equivalent to the radiation hazard of the cumulative release of all other radionuclides during the accident.

Release of Po from lead may depend on formation of the intermetallic compound Po-Pb, occurrence of oxide films on the lead surface and on other factors that are yet to be investigated. They were not considered in this analysis.

The estimated release of radioactivity in the postulated accident at BREST reactor corresponds roughly to Level 5 according to the international scale of events at nuclear plants (i.e. an accident involving risk to environment). Coolant purification to remove Po and other radionuclides would allow bringing the accident consequences down to Level 4 or even Level 3. However, if this accident develops to cause fuel damage, radioactive release may increase dramatically. This situation should be further investigated at subsequent design stages.

In a similar accident at sodium reactor, Na will burn at a rate of 1 to 3 t per hour. If 15 % of Na combustion products are carried over, release of ^{24}Na alone will exceed 10^5 Ci in four days (considering its decay) – and this is the minimum level, let alone ^{22}Na, ^{137}Cs and ^{131}I. Much sodium will burn out during this period. As a result, fuel will not be cooled any longer and will fail, which will cause a much greater release of radioactivity.

Given the worst weather conditions and disregarding precipitation from air, concentration of Na and its compounds in air at the distance of 10 km from the site may reach 10 mg/m^3, which is equivalent to 100÷1000 MPC in terms of Na chemical toxicity (for Na compounds MPC = 0.01 – 0.1 mg/m^3).

Thus, the analysis showed that release of radioactive and toxic materials in the accident involving failure of external barriers, will be three times greater in sodium reactors than in lead-cooled reactors because of sodium burning. This estimation does not take into account the consequences associated with loss of fuel cooling and its subsequent failure in the course of Na burning.

3. RESEARCH IN PHYSICS

Experimental and computational studies on the BREST-OD-300 reactor physics are largely aimed at:

- setting up a system of constants and software properly verified and certified by the Russian Regulatory Authority;
- demonstrating the degree of reliability and accuracy in calculating the main neutronic characteristics of the BREST-OD-300 core;
- proving the nuclear and radiation safety of the BREST-OD-300 facility and its on-site fuel cycle;
- adjusting and refining the design parameters of control systems, first and subsequent critical charges, refuelling programmes.

An international test model of the BREST-OD-300 reactor was developed for verification purposes. It appears sensible to have its characteristics analysed by as many Russian and foreign experts as possible with the use of various neutronic codes and nuclear data. The international scientific community is invited to take part in such testing. A test model for verification of transient and accident analysis codes is under development at present.

Validation of nuclear data and computer codes relies on the following integral experiments:

- measurement of removal cross-sections, determining the number of in-pile fast fissions, which allows judging the accuracy of description of the inelastic scattering process at 0.8–10.5 MeV;
- a series of critical experiments on spherical models with uranium and plutonium fuel and with a lead reflector of varied thickness;
- a series of experiments at the ROMB facility with "pancakes" of highly enriched uranium and lead,
- a series of experiments at BFS-1 and BFS-2 facilities in NRC "FEI".

Of special importance among those is the work with the BFS facilities, which was carried out under a special extensive programme for experimental investigation of the BREST-OD-300 neutronics and for reactor modelling [5].

The assessed accuracy of calculating the main neutronic parameters of BREST-OD-300 is an integral numerical value which reflects the progress made in understanding the reactor physics. Such values are given in Table II with reference to certain key dates and in comparison with the tentative requirements. The year 1990, when macroscopic experiments were practically nonexistent, was taken as a starting point for the analysis. The recent situation (2002) is characterised by several good benchmarks and an experiment on a part-

scope model of this reactor. As seen from the table, perceptible progress has been made in respect to many key parameters, with convergence of required and actual accuracies. According to current plans, at least the first phase of experiments on a full-scope BREST-OD-300 model at the BFS facility is to be completed in 2004. Assessments show that, given today's errors of experimental procedures, the required accuracy is likely to be attained for some parameters, such as criticality and power density distribution. Not so optimistic is the prospect for such parameters as the breeding ratio, some reactivity effects, and the worth of controls.

While developing the demonstration reactor BREST-OD-300, its designers realise that it is only pilot operation of the reactor that will allow reaching the required accuracy in calculating its physics, which is essential for description of reactivity variations in the course of establishing an equilibrium fuel composition and during subsequent operation in the equilibrium mode. Considering the similarity of fuel composition in the demonstration and commercial BREST reactors, the results of pilot operation may be assimilated in the commercial reactor design, using the perturbation theory methods.

Table II. The required and achieved accuracy in calculating the main neutronic characteristics of BREST-OD-300 reactor

Parameter	Required	Achieved		
		1990	2002	2004
K_{eff}	0.5%	2.5%	1.0%	+
Breeding ratio	0.02	0.06	0.04	?
CPS controls' worth	5%	30%	20%	?
Power density distribution	2%	5%	3%	+
Void effect of reactivity	0.2%	1.1%	0.4%	+
Doppler	10%	20%	15%	+

4. BREST-OD-300 FUEL TESTS IN BOR-60 REACTOR

In 1998-2001, a loop channel (or, in fact, a loop-in-channel construction) was designed and manufactured for testing BREST fuel rods in the BOR-60 reactor. This facility has a fuel assembly with BREST reactor fuel rods which differ from the real components only in length. The loop channel is fitted with all the components of the BREST reactor, such as fuel assembly, circulation circuit, oxygen monitoring and maintenance instruments, system for removal of impurities from the circuit, temperature monitoring and cladding failure detection devices. In the process of the loop channel manufacture, various technologies were tried out and perfected, including fabrication of U-Pu mononitride fuel [6], fabrication of the fuel cladding from EP-823 steel at the factory, lead injection, sealing, monitoring, the processes of fitting together the fuel assembly and the channel itself. Between January and May of 2002, the loop channel spent 2500 hours in the reactor and the fuel burnup reached 0.44%. Today, the channel and fuel rod materials are undergoing postoperational studies. Visual examination of the channel and the fuel rods after their cutting showed the steels to be free of corrosion. The design, manufacture and testing of the channel and fuel rods are described in detail in Ref. [7].

5. THERMAL-HYDRAULIC STUDY OF THE CORE

The work on the BREST-OD-300 design called for experimental studies on the thermal hydraulics of the core [8]. Considering the low coefficients of heat transfer in the lead coolant as compared to sodium (e.g. in BN reactors) and the lack of practical knowledge on the square fuel lattice employed in these reactors, investigations were carried out to determine the effect made on heat transfer coefficients by Peclet number (Pe), fuel rod pitch (s/d), spacer grids, radial and axial variations in power density, and by other factors typical of BREST reactors. Serious attention was paid in the studies to variations in the temperature of fuel rods in the regular lattice and of those located at the boundaries of the core regions with different fuel rod diameters and heat rates.

Experimental investigation of heat transfer coefficients and fuel rod temperature distributions was performed with the use of *thermal-hydraulic models* of the same design but differing in the mockup fuel rod pitch (s/d=1.46; 1.28 and 1.25), as well as in the absence or presence of spacer grids. The models were made as assemblies of 25 mockup fuel rods in a square lattice, placed in a rectangular shell. At the central mockup rod, of rotary design, surface temperature measurements were taken along its perimeter and length by means of micro-thermocouples caulked in the surface or moving along the rod. Coolant temperature was measured in all cells at the assembly outlet.

The *coolant was simulated* by a eutectic sodium-potassium alloy (22% Na + 78% K), with the Prandtl number numerically close to that of lead. Thereby close similarity of heat exchange processes at the "fuel rod – coolant" interface was provided, assuming certain "purity" of the coolants under consideration and absence of thermal-chemical phenomena on the heat exchange surface.

Thermal modelling of fuel rods (fissile material – uranium or plutonium mononitride, cladding – stainless steel, bond – lead) was fairly rigorous (with an accuracy of 5 %) using the fourth harmonic of temperature field expansion into Fourier series ($\kappa_0 = 4$), which is basic to the regular square fuel lattice. The experimental results are fully described in Ref. [8].

6. STEEL CORROSION TESTING

The problem of steel resistance to corrosion in lead is tackled with the use of a special lead coolant technology which allows forming protective oxide films on steel surfaces and keeping their thickness within optimal limits. This technology was first applied at facilities using Pb–Bi eutectic. The operating experience of these facilities was instrumental in choosing structural materials for the reactor under development, with this choice subsequently justified by long-term tests in lead under near-operational conditions in terms of lead temperature flow velocity. Steels of various classes were tested: e.g. austenitic, pearlitic and ferritic-martensitic steels with 9 to 12 % of Cr.

Apart from compatibility with liquid lead, the choice of materials was also guided by other criteria, such as high-temperature strength, manufacturability, resistance to radiation (for core materials) and to corrosion in high-parameter steam (for steam generator materials).

As a result, the structural materials recommended for the BREST-OD-300 reactor were assessed for corrosion resistance on the strength of 13500-hour tests [9].

With the optimal coolant oxygen regime identified, it is now one of the parameters involved in corrosion fatigue tests of candidate steels and their welds (tests have gone for 6,000 hours, of the 20,000 hours planned).

First studies have been carried out to test the steel for durability in liquid lead at 550°C. Liquid lead has not been found to have any significant effect on the long-term strength of the steels. Its reduction in lead is no greater than 10-13%.

The tested prototype pumps have proved their serviceability over a stretch of 3000 hours. Specifications were developed for the semifinished items required for the reactor, and their production batches have already been received. Welding methods have been developed for some reactor components.

Work on demonstrating the performance of the selected steels and their welds is being done today in the following main areas:

- study of steel corrodibility in liquid lead, lead vapour, at the liquid lead – inert gas interface, and in water vapour;
- investigation of mechanical properties of steels exposed to liquid lead;
- investigation of the irradiation effect on the structure, physical, mechanical and corrosion properties of steels in liquid lead;
- development of technologies for manufacture, bending and welding of semifinished products; quality assessment of the semifinished products and welds

7. COOLANT TECHNOLOGY JUSTIFICATION

The ever-increasing requirements for the safety and reliable performance of nuclear reactors spur up the search for new coolants which will have advantages over traditional fluids, such as water, sodium, etc.

One of such coolants is liquid lead. In its physics and chemistry, liquid lead is rather similar to the lead-bismuth eutectic. A great body of data has been amassed on the physical, chemical, thermal and other properties of this alloy. Procedures and experimental capabilities are available and are actually being used for justifying the choice of lead as a coolant for power reactors, of which BREST-OD-300 is the forerunner.

Lead dissolves many chemical elements and compounds, including some components of structural materials, with the ensuing risk of structural deterioration and loss of circuit integrity. Material dissolution (corrosion) may be effectively slowed down by protective iron and chromium oxide films on steel surfaces. Liquid lead reacts perceptibly with oxygen, giving rise to slag (containing oxides of the coolant itself, of structural steel components, etc.), which may deposit on the circuit surfaces, impairing its thermal-hydraulic characteristics.

Oxygen content in the coolant and in the circuit as a whole is a highly important factor in normal operation of the circuit. Oxygen excess leads to slagging, whereas its deficiency is likely to cause dissociation of protective oxide coats on structural materials and development of corrosion processes. Therefore successful operation of BREST-OD-300 will depend on properly controlled coolant quality, with the content of impurities kept at an optimal level (e.g. of oxygen, oxide compositions based on structural materials, etc.).

Provisions should be also made to prevent fouling of the gas plenum components and to clean the gas circulating in the reactor's gas plenum from the products of lead evaporation, material corrosion, and other impurities.

All these problems are addressed in implementing the lead coolant technology. The term "coolant technology" implies a package of organisational and technical work, processes and their associated systems (facilities), all aimed at providing the required cleanness of the circuit and corrosion resistance of its structural components during construction, startup, maintenance and operation of an experimental rig or a full-fledged reactor facility.

Computations and experiments [10] resulted in the choice of the following lead coolant technologies accepted for further development:

— coolant and circuit treatment by hydrogen;
— coolant quality control by maintaining its oxygen content with the use of solid-phase oxidiser;
— removal of solid impurities from the coolant by means of filters;
— gas cleaning from suspended particles and aerosols by means of filters.

The information obtained to date from studies and experiments is sufficient for developing a draft procedure for lead coolant management.

8. FUEL CYCLE OF BREST-OD-300

The BREST-OD-300 project includes a nuclear power plant with a demonstration 300 MWe liquid metal reactor BREST, an on-site fuel cycle and a complex for treatment and storage of radioactive waste. The design studies undertaken to prove feasibility of BREST reactors of different power (600 and 1200 MWe) to be used in the prospective large-scale nuclear power, followed the same philosophy as that of the 300 MWe reactor.

The nuclear fuel cycle of BREST-OD-300 [11] affords practically unlimited availability of fuel resources for nuclear power, owing to recycling of U-Pu fuel with equilibrium composition (CBR~1), so that only small amounts of depleted or natural uranium need to be added to it. Moreover, such a cycle allows attaining radiation equivalence with allowance for migration. With radiation equivalence, waste sent to disposal, will have activity and nuclide composition such that the temperature and stability of material subject to disposal and the risk of nuclide migration, considering their biological hazard, are similar to those of natural uranium deposits.

The closed on-site fuel cycle of the BREST-OD-300 plant was designed to provide fabrication of 14 t of (U, Pu)N fuel per year, including ~3.5 t (U, Pu)N/year for BREST reactor and another 10.5 t (U, Pu)N/year for BN-800 under construction at the Beloyarsk site. The on-site facilities were also designed to provide fabrication of 49.1 t of (U, Pu)N fuel for the first cores of BREST and BN-800 reactors (17.6 t and 31.5 t, respectively). The on-site fuel cycle consists of the following fabrication-related facilities:

— production of Pu mononitride;
— FA cutting and stripping of fuel rods;
— reprocessing;
— preparation of press powder and fuel fabrication;
— preparation of claddings and fuel rod components;
— fabrication of fuel rods;
— fabrication of fuel assemblies.

In the BREST-OD-300 project, the emphasis is on engineering rather than administrative nonproliferation provisions. This is proved by the following design features:

— there is no blanket in BREST-OD-300. Any specially fabricated Pu-breeding fuel assembly placed in the core under equilibrium operation, will cause introduction of considerable negative reactivity, which will be inevitably detected during reactor startup after refuelling;
— the BREST fuel cycle does not include shipment of spent fuel assemblies to a reprocessing plant. Spent fuel assemblies are to be sent after a year of cooling to on-site cycle facilities via a transport corridor connecting the latter to the reactor section. This helps avoid all risks and costs associated with fuel shipment for reprocessing and obviates the need for the associated transportation and handling gear;
— BREST fuel - before or after reprocessing - cannot be used for production of nuclear charges without a proper separation facility. A crucial requirement for reprocessing technique is to keep uranium and plutonium inseparable and their proportion in fuel constant at all reprocessing stages;
— fuel reprocessing and fabrication takes place in unattended heavily shielded cells;
— reprocessed fuel contains up to 1% of fission products, which simplifies safeguarding of nuclear material. (This feature of BREST fuel is sometimes referred to as "inherent safeguards").

9. EXPERIMENTAL CAPABILITIES FOR R&D IN SUPPORT OF THE BREST TECHNOLOGY

The work on design of the demonstration plant with a 300 MW lead-cooled fast reactor (BREST-OD-300) and on-site fuel cycle at Beloyarsk has progressed to a level where the experimental studies for design validation need to be expanded.

In 1999-2002, the participating organisations (NIKIET, FEI, VNIINM, NIIAR, VTI, CNIITMASH, CNII KM "Prometei", OKB "Gidropress", "Gidromash, and others) launched project justification studies using the available experimental facilities and focusing on the flowing areas [12]:

- corrosion studies and qualification of structural materials for the core and reactor internals (FEI, CNII KM "Prometei", NIKIET, CNIITMASH);
- neutronic studies (FEI);
- thermal-hydraulic studies (FEI);
- research in coolant technology (FEI);
- in-pile tests (NIIAR, Sverdlovsk Branch of NIKIET)

10. CONCLUSION

A nuclear generating complex has been designed, including a demonstration plant with BREST-OD-300 reactor, on-site nuclear fuel cycle and radwaste treatment facilities, to be built on the Beloyarsk NPP site. The experiments planned for the complex are expected to confirm the feasibility of building a large-scale nuclear power industry based on fast reactors. Full-scope R&D should be continued to have the design comprehensively validated.

Ample design documentation has been produced to obtain a license for the site, to start review procedures (governmental, environmental, regulatory, etc.) and design refinements so that a construction license may be obtained in a 3 or 4 years' time.

The cost of the BREST Project, comprising the plant proper, the fuel cycle and the radwaste treatment facility to cater for the BREST-OD-300 and the BN-800 under construction on the same Beloyarsk site, is estimated at US$ 825 million, including:

Capital cost of the NPP	285 M$
Capital cost of the NFC and RW complex	355 M$
Project R&D	185 M$

The nuclear generating complex built to this design will have nuclear fuel delivered to it only once, to make the first core, while its radioactive waste, treated and cooled for ~ 150 years, will have a radiotoxicity equivalent to that of mined uranium and therefore may be buried without upsetting the natural radiation balance.

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SVBR-75/100 — LEAD-BISMUTH COOLED SMALL POWER MODULARFAST REACTOR FOR MULTI-PURPOSE USAGE

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Abstract. Today's nuclear power is in the state of an intrinsic conflict between economic and safety requirements. This fact makes difficult its steady development. One of the ways of finding the solution to the problem is use of modular fast reactors cooled by lead-bismuth coolant that has been mastered in conditions of operating reactors of Russian nuclear submarines. Based on this experience, the small power fast reactor for multi-purpose usage (SVBR-75/100) has been developed. The small power reactors make it possible to fabricate the whole reactor at the factory and deliver it to the NPP site in practical readiness by using any kind of transport including the railway. Reactor installation SVBR-75/100 was designed in compliance with a conservative approach. This approach presumes to use to the maximal extent the technical solutions and temperature parameters, which have already been verified by operating experience. Further, when a one-through steam-generator producing over-heated steam is used, technical and economical parameters will be considerably improved. Technological maintenance of fissile materials non-proliferation has been ensured at all lifetime stages. Refuelling in developing countries has not been provided. On ending the lifetime (~10 years), the reactor will be transported to the Supplier-country along with the core in the "frozen" coolant.

1. INTRODUCTION

To achieve the mature phase of nuclear power (NP) development with replacement of approximately 50 % of fossil fuel sources for electricity production, it is projected that several nuclear power technologies (NPT) including the nuclear power plants (NPP) of a certain type and a corresponding nuclear fuel cycle (NFC) will be superseded.

The NPT that best meets the requirements of a current stage of NP development in a certain country will dominate at each stage of NP development.

Probably, the duration of each stage necessary to achieve the mature phase of the NP would take many decades, which is caused by significant sluggishness of development of any new NPT.

The mature phase of the NP will be featured by:

— Domination of fast reactors (FR) operating in the entirely closed NFC;
— The most complete realization of the inherent safety principles;

— Finding the practical solution to the problem of handling long-lived radioactive waste (RAW);
— Maximum of technological maintenance of nuclear fissile materials (NFM) non-proliferation.

As the alternative energy sources exist together with the NPPs, at each stage of NP development in conditions of a liberalized electricity market, the NPPs must be competitive with the heat power plants (HPP) using fossil fuel.

The NPT will be superseded by the other one in case:

— The new NPT showing the better technical parameters appears;
— The NPP competitiveness is lost due to the higher paces of increase of the nuclear fuel costs at exhaustion of cheap nature uranium sources comparing with the paces of increase of the fossil fuel costs (that is low probable) or due to the higher costs of the NPPs and NFC caused by imposing the more stringent requirements.

For example, if the regulation authorities specify the requirements for non-proliferation technological maintenance as well as high safety requirements, the NPT competitiveness would be noticeably affected.

At the same time, increase of the cost of any source used by a certain NPT will stimulate both the exploration and development of this source (e.g. uranium) and launching works on changing over to the new NPT using this source more effectively (e.g. NFC closing) or using the new source (e.g. thorium).

However, at the electricity market the NPP competitiveness with HPPs using fossil fuel is not sufficient enough to ensure self-financing of NP development with a pace providing increase of the NP share in the total production of electricity generated by all kinds of power sources.

This is caused by the fact that at the liberalized electricity market in the developed countries the rate of electricity costs is decreasing due to excess of generating power capacities. In Russia the investment potentials of the NP have been limited by regulated cost rates, increase of an annual electricity production cost, and a value of total annual electricity production and they cannot cover the long-term investment needs consisting of the costs for:

— Increasing the loading factor (LF) of the NPP;
— Enhancing safety of the first generation units;
— Extending the life time of the units over the designed ones;
— Constructing the units of a high and medium extent of readiness;
— Decommissioning the NPP units with an expired lifetime;
— Constructing the new replacing power capacities compensating for the decommissioned units;
— Constructing the new NPPs providing a desired pace of NP development.

Along with it, in conditions of a market economy the NP cannot rely on any noticeable support of the state financing.

This fact put forward the requirement for the NPP competitiveness at the market investments (including those based on the credit repayment).

Realization of this requirement needs for considerable reduction of specific capital costs of NPP constructing and construction terms making these parameters close to those of the modern steam-gas HPPs. Besides, to reduce the investor's risk, it is necessary to considerably improve the safety level in order to eliminate the severe accidents like Chernobyl one. This is also necessary for ensuring an acceptance of the NP by public opinion when its scale is considerably increased.

It is very difficult to solve this problem on the basis of evolutionary improvement of the traditional NPP designs with thermal neutron reactors because an intrinsic conflict between economic and safety requirements is peculiar to such reactor installations (RI). This cause a necessity of constant increasing the unit power of the reactors that results in increasing the total investments, increasing the construction terms and reducing an investment attractiveness of the design.

Besides, the existing thermal neutron reactors cannot provide long (hundreds of years) development and functioning of the NP due to the low efficiency of using natural uranium power potential even in the closed NFC that makes electricity production more expensive at low prices of natural uranium.

The solution to this problem can be find by using an innovative NPT that uses a new type of the fast reactors (FR) which must not build up plutonium with a short doubling time (this task has lost its actuality).

This enables chemically inert heavy liquid-metal coolants (HLMC) (instead of sodium) with high boiling point to be used for heat removal, i.e.: eutectic lead-bismuth alloy (45 % of Pb, 55 % of Bi) that has been mastered in conditions of operating the nuclear submarines' (NS) reactors of Russian Navy [1] and currently mastering lead coolant.

Available reference information on explored bismuth resources has not allowed use of lead-bismuth coolant (LBC) in the large scale NP. However, just recently the specialized MINATOM enterprises – OAO “Atomredmedzoloto” and VNIPI of industrial technology – have carried out technical and economical investigations into an opportunity to organize large scale bismuth production in Russia and estimations of bismuth sources in the Commonwealth of Independent States (CIS). On the basis of the explored bismuth mines of the only Chita region in Russia, it is possible to produce bismuth in quantities sufficient enough to put n operation ~ 70 GWe of NPPs with LBC cooled FRs. In addition, there are large bismuth sources in the North Caucasus. It is possible to put in operation ~ 300 GWe by using the bismuth mines of Kazakhstan.

Japanese explorers have determined the world's available bismuth sources to be ~ 5 million tons [2].

It should be highlighted that in compliance with a general geological and economical law, the quantity of the mineral raw ore increases as the square of the cost that the consumer would be ready to pay.

At existing costs of bismuth in the world its contribution to the capital costs of constructing the large NPP on the basis of considered FRs is ~ 1 %. For that reason, in practice the real technical and economical parameters of the NPP will not be noticeably worth even in case of the bismuth cost increases several times.

In the future when cheap bismuth sources have been expired, it will be possible to change over to lead-bismuth alloy of a non-eutectic composition with reduced bismuth content and the higher boiling point. For example, when bismuth content in the alloy is reduced 5.5 times, the melting point is increasing from 125 to 250 °C that is considerably low than a melting point of pure lead.

This report is considering the NPT based on using small power lead-bismuth cooled FRs SVBR-75/100 (**L**ead-**B**ismuth **F**ast **R**eactor of equivalent electric power **75 ... 100** MWe depending on the steam parameters). In the nearest 15 ... 20 years it can be implemented both in developed and developing countries with meeting the most requirements to the Generation IV DOE reactor systems and International Project INPRO. Due to the higher technical and economical parameters of the NPP and the higher safety level [3], this technology can be considered as one of the possible ways of gradual replacement of the current NPT based on using the light water reactors (LWR).

An effect of improving the economic parameters of the NPPs based on RIs SVBR-75/100 is achieved due to lack of many safety systems necessary for the NPPs with LWRs, which make NPPs of this type considerably more expensive.

As it can be seen further, at the minimal starting costs of industrial mastering that NPT in the process of its evolutionary improvement, all these enable to implement gradual meeting of all requirements, which are peculiar to the mature phase of NP development.

2. BRIEF DESCRIPTION OF EXPERIENCE OF LBC USAGE

In the early 1950s, nearly at the same time the USA and the USSR launched their development programs on RIs for NSs. Both countries developed two types of RIs: pressurized water reactors and reactors cooled by liquid-metal coolants (LMC).

In the USA sodium was selected as LMC because its thermo-physical characteristics were better than those of LBC. The ground-based test facility-prototype of the RI and experimental NS "Seawolf" were constructed. However, operating experience revealed that option for the coolant, which was fire- and explosion-dangerous in contact with air and water, did not prove itself. After several RI accidents occurred at this NS it was decommissioned together with the compartment and replaced by a pressurized-water RI. R&D works on mastering LBC were also carried out in the USA. However, a selected approach of finding the solution to the problem of structural materials corrosion resistance, control and coolant quality maintenance (coolant technology) did not give any positive results, and the works were stopped.

In the USSR lead-bismuth eutectic alloy was selected as LMC in the very beginning. For fifteen years the certain organizations had been carrying out these works under IPPE scientific supervision. As a result, the problem was solved successfully, and it was verified by further many-year experience of RIs operating at the NSs. When operating the second generation RIs, there were no problems caused by structural materials corrosion in the primary circuit and violating the circuit purity standards [4].

The problem of ensuring radiation safety that was caused by forming polonium-210 was solved in the same way. During the whole period of operating LBC cooled RIs, including the primary circuit equipment's repair period and removal of spilled LBC, there were no cases of personnel's extra-irradiation over the permissible limits in terms of this radionuclide.

Altogether eight NSs with LBC cooled RIs were constructed. The first experimental NS of Project 645 had two reactors. Each of the other seven NSs of Project 705 (in terms of NATO – "Alpha") had one reactor. Due to its speed parameters this NS was entered into Guinness Book of Records.

Besides, two full-scale ground reactor facilities-prototypes were constructed and operated in IPPE (Obninsk) and NITI (Sosnovy Bor). A total sum of operating time of the considered type RIs has been ~ 80 reactor-years [1]. The new nuclear power technology that has no analogues in the world has been demonstrated in industry. Currently the conditions for introducing this technology into the civilian nuclear power have been formed.

3. BASIC STATEMENTS OF RI SVBR-75/100 CONCEPT

RI SVBR-75/100 was designed mainly in compliance with a conservative approach. This approach allows: without exceeding the limits of the experimentally tested mode parameters of the primary and secondary circuits, to use to the maximal extent the already mastered fuel and structural materials and verified in practice the principal technical solutions to the equipment components and RI scheme.

This approach ensures a high extent of succession of the RI SVBR-75/100 technical solutions, first of all, the technical solutions of LBC cooled NSs' RIs that has been favoured by nearness of their scale factors. Adhering to this approach reduces the execution terms, R&D scopes and costs, investment risk, ensures reliability of the RI and its operation safety. These factors make it possible to avoid the errors typical of an initial stage of mastering the innovative NPT.

Use of the conservative approach does not mean that the new technical solutions should not be used and an evolutionary way of NP development should be only followed. This would cause stagnation and hindrance of the scientific and technical progress. However, use of verified in practice technical solutions ensures the applicable technical and economical parameters of the NPP [3]. For that reason, the new, perspective technical solutions that considerably improve the parameters of the RI will be used when changing over to the next generation of the given type RIs after carrying out the necessary R&D.

Expediency of this approach has resulted from the analysis of the technique development history. It shows that for successful introduction of new technologies, the share of new technical solutions in complicated systems should not be too high. Ignorance of this fact can result into considerable delay of start launching, unnecessary over-expenditure of materials and financing. For that reason, when RI SVBR-75/100 was being developed, priority was given to the already developed technical solutions even if they did not ensured achievement of the highest technical and economical parameters.

With due account of all mentioned, the following basic approaches and technical solutions have been realized in the RI SVBR-75/100 design:

(i) A monoblock (integral) design of a pool type is used for the primary circuit equipment. Valves and LBC pipelines are completely eliminated;
(ii) A two-circuit scheme of heat removal is used;
(iii) The levels of coolants' natural circulation (NC) in the heat-removal circuits are sufficient enough to ensure reactor's heat decay removal without dangerous over-heating of the core;

(iv) A reactor monoblock with a safeguard vessel is installed and fixed in the tank of the passive heat removal system (PHRS). The tank is filled with water and also performs the neutron protection function;

(v) A wrappless design of the fuel sub-assemblies (FSA) is used. This ensures high cross heat-mass-exchange in the core and eliminates unallowable over-heating of fuel elements at large blockages of flow rate at the core inlet;

(vi) A steam-generator (SG) operating in compliance with a multiple NC scheme and producing saturated steam is used. This ensures the best lifetime and operating parameters, e.g. reliable RI operation at any power levels, simplicity of maintaining LBC in a liquid state at low power levels (including the mode of heat decay removal via the SG);

(vii) A slow-rotating gas-tight uncontrolled electric engine of the main circulation pump (MCP), which power does not exceed 500 kW, is used. This eliminates the necessity to seal the rotating shafts, enables to use the ball-bearings with greasing and provides the necessary against-cavitation condition at the suction of the MCP impeller due to coolant column's hydrostatic pressure;

(viii) The RI equipment can be repaired or replaced;

(ix) On ending the lifetime, refuelling can be performed at once, FSA-by-FSA;

(x) It is possible to use different kinds of fuel (UO_2, MOX fuel with weapon or reactor Pu, TRUOX fuel, nitride fuel) without changing the reactor design and at meeting the safety requirements.

With due account of the relatively high cost of LBC, there have been developed the measures reducing the specific mass of LBC in the RI.

The summarized analysis of experience of developing RIs of different power capacities [5] has revealed the LBC specific mass decreases at reducing the RI nominal power.

Along with this, reducing the LBC specific mass is limited. It is caused by the fact that at small dimensions of the core, it is impossible to provide core breeding ratio (CBR) ≥ 1. Computations have revealed that an optimal diameter of the core should be not less than $1600 \div 1700$ mm at height 900 mm. These core dimensions make it possible to achieve equivalent electric power of the reactor ~ 100 MWe. In this case, CBR $\cong 1$ is provided not only for the mixed nitride fuel but also for the less dense but well mastered MOX fuel. This point can be carried out if the volumetric fuel fraction is not lower than $55 \div 60$ %.

Reduction of the LBC specific mass in small-sized FRs, for which the core power density is several times less than that of sodium cooled FRs, is also achieved by elimination of the in-reactor storage of spent nuclear fuel (SNF) and in-reactor refuelling mechanisms (rotating plugs, etc.).

In this case, refuelling is performed once during the core lifetime. For that purpose, a special refuelling equipment is used, it is also used for refuelling all reactors of the power unit. The refuelling technology is similar to that of LBC cooled NSs' RIs.

Another way of reducing the LBC specific mass is increasing its average velocity in the RI and diminishing the length of the LBC circulation circuit. However, this way has its own constraints caused by the necessity to provide the safety requirements.

The first requirement is caused by the necessity to provide the power level of the reactor with naturally circulating LBC to be not less than 5 % of N_{nom}. This makes it possible to eliminate dangerous temperature increase in case of shutting down the MCPs.

The second requirement is caused by the necessity to provide an effective separation of steam bubbles from LBC with steam surfacing to the LBC free levels in case of an accident with leaking SG tubes. This is necessary for elimination of steam ingress into the core and impermissible pressure increase in the monoblock vessel.

The necessity to satisfy the highlighted requirements resulted into development of the LBC circulation scheme in which core hydraulic resistance equals to 90 % of the total hydraulic resistance of the primary circuit and hydraulic resistance of the SGs, in which LBC flow rate is much less, only equals to 10 %.

With due account of the highlighted requirements, the specific mass of bismuth in RI SVBR-75/100 is ~ 1100 t/GWe.

It should be highlighted that the low values (25 ... 30 %) of the LBC volumetric fraction in the core ("tight" lattice of fuel elements) and LBC specific mass do not deteriorate the safety parameters of RIs SVBR-75/100 in cases of shutting down the MCP and leaking SG tubes (as computations have revealed) but in the case of unauthorized insertion of positive reactivity as well. The latter is caused by a sufficiently high negative feedback being typical of small power reactors and a low time of delaying its temperature component at the LBC core inlet (extending of the lower core-plate) coupled with sufficient heat-accumulation ability of the monoblock.

The following have been provided at the selected power level (100 MWe):

— The lifetime duration is ~ 53000 eff. hours if mastered oxide uranium fuel is used (CBR = 0.87);
— CBR ≥ 1 if MOX fuel is used, the reactor operates in the closed fuel cycle in the mode of fuel self-providing;
— CBR ≥ 1 if mixed nitride fuel is used, the reactor operates in the mode of fuel self-providing at a burn-up reactivity margin being less than β_{eff} or in the mode of extended breeding with CBR = 1.13 at a plutonium doubling time being ~ 45 years;
— A burn-up reactivity margin is less than β_{eff}, the lifetime duration is ~ 80000 eff. hours in case of using uranium nitride fuel;
— Reactor's heat decay removal is entirely passive, heat is removed through the monoblock vessel to the PHRS tank;
— Complete plant fabrication of the reactor monoblock, RIs are produced in large quantities that improve the quality of works and reduce the cost;
— The reactor monoblock can be transported by railway, truck or marine transport (with fuel in a nuclear and radiation-safe state due to LBC "freezing" in the monoblock vessel that also meets non-proliferation requirements);
— The term of constructing the NPP unit can be considerably reduced as modules are delivered in high plant readiness and the assembling scopes are sharply reduced. (This improves the terms of receiving the NPP construction credits and reduces the period of capital investments recoupment);
— The NPP unit in which the RIs have been replaced by the new ones can be renovated in 50 ... 60 years. This postpones the necessity to construct the replacing power capacities to 50 years;
— The cost of decommissioning the unit can be considerably reduced as after removing the monoblock, no radioactive materials remain in the main reactor building;

The NPP units with LWRs which RIs have exhausted their reactor lifetime can be renovated by installing the necessary number of RI SVBR-75/100 in the empty SG and MCP rooms.

The basic parameters of RI SVBR-75/100, a longitudinal section of the reactor monoblock and reactor compartment are cited in Table I, Fig. 1, Fig. 2.

Table I. Basic parameters of RI SVBR-75/100

	Name and Dimensions of the Parameter	Value
1	Heat power (nominal), (MW)	280
2	Steam-production, (t/h)	580
3	Pressure of generated saturated steam, (MPa)	9.5
4	Feed water temperature, (°C)	240.9
5	Primary circuit coolant's flow rate, (kg/s)	11760
6	*Primary circuit coolant's temperature, outlet/inlet, (°C)*	482/320
7	Core dimensions: diameter × height, (m)	1.645 × 0.9
8	The number of fuel elements	12114
9	The number of CPS rods	37
10	Average power density of the core, (kW/dm^3)	146
11	Average linear load of the fuel element, (kW/m)	~24.3
12	The time interval between refuellings, (years)	~8
13	Uranium fuel (UO_2) load: mass, (kg)/enrichment, (%)	9144/16.1
14	The number of SG modules	2 × 6
15	The number of MCPs	2
16	Power and head of the MCP, (kW/MPa)	450/0.55
17	The core lifetime, (eff. hours)	53000
18	LBC volume in the primary circuit, (m^3)	18
19	Dimensions of the reactor monoblock: diameter × height, (m)	4.53×7.55

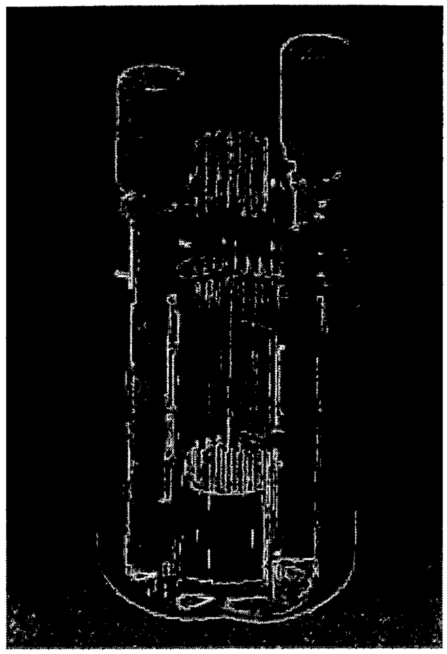

FIG. 1. Reactor monobloc.

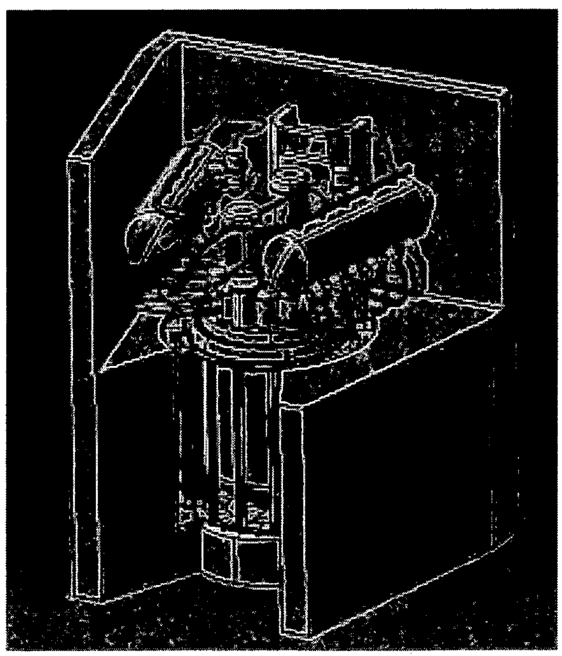

FIG. 2. Reactor compartment.

4. SAFETY PROVIDING CONCEPT

Lead and bismuth natural properties, physical features of FRs coupled with an integral (monoblock) design of the primary circuit equipment make it possible to eliminate deterministically an opportunity of the certain severe accidents.

High boiling point of coolant enhances reliability of heat removal from the core and safety due to lack of the heat removal crisis phenomenon and being coupled with a safe-guard vessel eliminates the accidents of the LOCA type.

Low pressure in the primary circuit enables to reduce the thickness of the monoblock vessel walls and reduce the limitations imposed on the temperature change rate in compliance with the thermo-cycling strength conditions.

LBC reacts with water and air very slightly. Development of the accident processes caused by primary circuit's tightness failure and SG intercircuit leaks occurs without hydrogen release and any exothermic reactions. There are no materials within the core and RI that release hydrogen as a result of thermal and radiation effects and chemical reactions with coolant. Therefore, the likelihood of chemical explosions and fires as internal events is virtually eliminated.

In the case of failure of all active cool-down systems and total blacking out the unit, elimination of core melting caused by residual heat release and keeping the monoblock vessel intact are ensured by an entirely passive way due to heat accumulation in the in-vessel structures and coolant and heat removal to the PHRS water tank through the monoblock vessel with further water evaporation. The "grace" period is about five days' time. A scheme of heat removal to the PHRS tank is shown in Figure 3 and Figure 4 shows how the maximal temperature of the fuel element's cladding and the water level in the PHRS tank depend on time.

Core melting is also eliminated at postulated LBC "freezing" in the SG. In this case, NC of LBC with a flow rate being ~1 % of the nominal one is performed over the continuously operated by-pass circuit past the SG from the central buffer chamber to the peripheral one via the holes in the shells, which have been provided for this purpose.

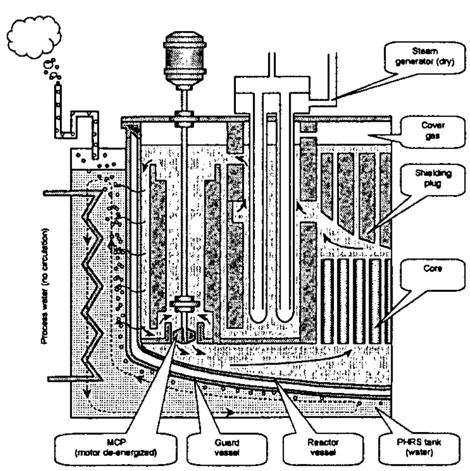

FIG. 3. Heat removal to the PHRS tank.

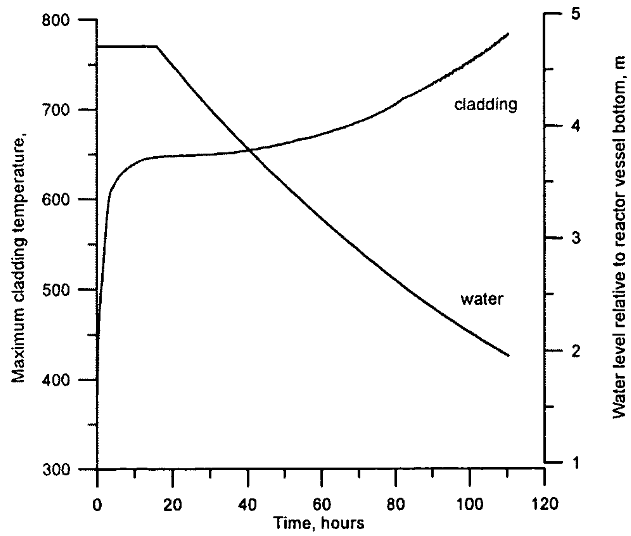

FIG. 4. Fuel element's cladding and the water level in the PHRS tank depend on time.

In case of unauthorized insertion of positive reactivity at postulated failure of all emergency protection (EP) drivers, elimination of prompt neutron reactor runaway is ensured by a special algorithm of compensating rods control, which is the part of the automatic control system. In this case, when the reactor operates at nominal power, during a certain time (~ 4 months) a reactivity margin controlled by an operator is much less than β_{eff}.

Besides, an efficiency of each rod is much less than β_{eff}, a rate of moving the absorbing rods extracted gradually is technically limited. For that reason, the inserted positive reactivity has time for being compensated by negative feedbacks without dangerous increase of the core temperature.

In the case of EP system failure caused by the external events not specified in the regulatory documents (for example, damage of all servo-drivers), there are fusible locks connecting a rod with a driver bar. When the coolant temperature exceeds 700 °C, EP rods that are installed in the "dry" channels are separated from the bars and drop into the core due to their gravity.

For considered fuel loads, the total void reactivity effect of the reactor is negative and the local positive void reactivity effect is less than β_{eff} and cannot be realized due to the coolant's very high boiling point and lack of the opportunity for gas or steam bubbles to arise in great quantities.

Elimination of water or steam penetration into the core caused by a large SG leak and consequent overpressurization of the monoblock vessel designed to be resistant against the maximum possible pressure under this condition are ensured by the coolant's circulation scheme. This scheme provides that steam bubbles are thrown out on the free coolant level by the moving up LBC flow. Then steam goes to the gas system condensers. In the event of their postulated failure, steam goes to the bubbler (PHRS tank) through the bursting membranes (see Fig. 5).

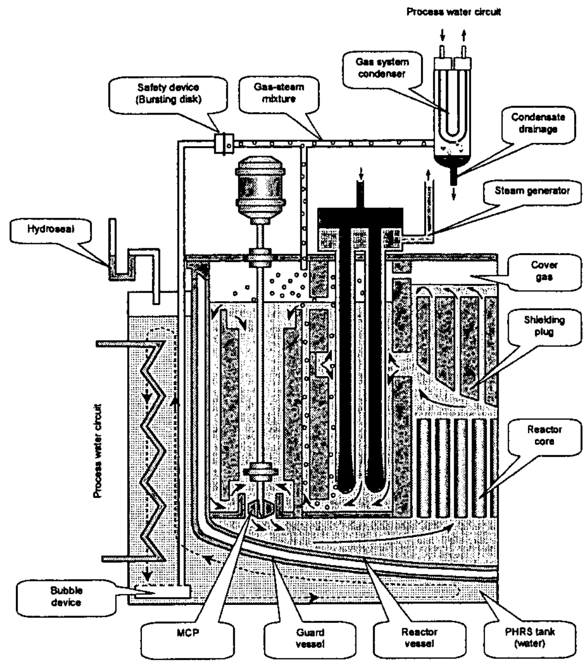

FIG. 5. Heat removal to the PHRS tank in case of failure.

Carried out studies have revealed that no equipment failures, personnel's errors or their combinations may cause core melting. Negative reactivity feedbacks ensure power decrease down to the value that does not cause core damage even in case of failure of all reactor shutdown systems and total blacking out.

The additional barriers of the safety providing system are the separate concrete cells of the RI (confinements) that restrict radioactivity release into the central reactor hall, and a protection shell of the central hall covering all RIs (containment) purposed to protect the reactor against external impacts. In an event of accident tightness failure of the primary circuit, high pressure radioactive exhausts (which can happen in LWRs) do not occur in LBC cooled RIs. For that reason, there is no need to design a containment of the unit and RI compartment to be resistant against high excess internal pressure. There is also no need to design a double containment with a water cooling system and a corium catch.

An extremely simple design of RI SVBR-75/100, lack of plant safety systems caused by developed inherent safety properties of the RI, that made it possible to couple the functions of RI safety systems with those of normal operating systems, sharply reduce the probability of personnel's errors. The consequences of any personnel's errors and their combinations do not affect the safety but only result in economical losses and the necessity to carry out unscheduled repair works.

RI safety does not depend on the equipment and systems of the turbine-installation. Major safety viable systems operate passively being independent on either right or wrong personnel's actions.

It should be highlighted that there are no valves or mechanical devices in the RI safety systems, which may cause failure of their operation in case of failure or switching off the

valves or mechanical devices that may be caused by someone's error or malicious actions or over-standard external impacts.

For that reason, RI safety systems' action has been assured by:

- Melting the locks of the EP rods and their free fall down to the core;
- LBC natural circulation, heat transfer via the main and safe-guard vessels, air convection in the gap and heat irradiation, water boiling in the PHRS tank in the modes of emergency heat decay removal;
- Rupture of the safety membrane that protects the monoblock from excess pressure at large SG leaks and failure of the gas system's steam condensers.

As computations have revealed, an extremely high safety potential typical of the considered RI is characterized by the following: even in an event of the postulated combination of such initial events as containment destruction, damage of the RI compartment overlapping and primary circuit gas system's serious failure with direct contact of the LBC surface with atmospheric air in the monoblock vessel that is possible in the case of terror attacks, neither reactor runaway, nor explosion, nor fire occurs, and the radioactivity release is less than that requiring the population evacuation.

Obtained results enable to conclude that the safety level of SVBR-75/100 reactors is higher than that of LWRs and sodium cooled FRs. It can be practically demonstrated at the stage of experimental operation of RI SVBR-75/100 with controlled simulation of different designed initial events and their combinations.

5. CONCEPT OF THE MODULAR NPP BASED ON RI SVBR-75/100

It is highlighted in [6] that "Modular plant fabrication of nuclear power systems and their assembling on the site will replace the existing expensive construction methods". Economical advantages of modular principle of constructing the NPP are also highlighted in [7]: "Measures on reducing the construction terms much affect the total capital costs especially at high record rates because in the course of construction, the credit payment may reach 25 percent and more of the total scope of investments. Modular production that makes it possible to fabricate and assemble the units at the plant but not on the site reduces the construction term and, consequently, expenses on the credit payment during the construction period".

Reduction of the investment cycle of constructing the NPP, that has been ensured by a modular structure of the NPP and delivery of ready fabricated modules, is extremely viable for the technical and economical parameters of the NPP to approach those of steam-gas HPPs with short investment cycles [8].

For developed countries, which power systems have high-voltage electric transfer lines with high transmission, it will be economically effective to use large modular power units. Maximal possible capacity of the modular type power-unit will not be restricted by maximal possible reactor capacity.

In case the large power modular power-unit is equipped with one turbine installation and at the existing technical level of turbine-constructing factories in Russia, the power-unit's capacity can be taken as 1600 ... 1800 MWe. SSC RF IPPE, FGUP EDO "Gidropress", FGUP "Atomenergoproekt" have developed a conceptual design of the two-unit NPP, which power unit includes the nuclear steam-supply system (NSSS) consisting of 16

RIs SVBR-75/100 (reactor modules) and one turbine-installation of 1600 MWe [3]. This allows to compare correctly the technical and economical parameters of that NPP to those of the NPP based on RI VVER-1500.

When NPP unit's capacity was selected, it was taken into account that specific capital costs of the reactor compartment ("nuclear island") would decrease at increasing the unit's capacity. It is caused by the fact that at increasing the number of modules in the NSSS, the cost of the equipment and providing systems installed beyond the RI compartments increases slightly. For that reason, their contribution to the specific capital costs of the reactor compartment will decrease.

Such systems and equipment include the refuelling equipment, coolant's in-taking equipment, equipment for coolant's transferring to the monoblocks at initial filling, etc. So, the specific capital cost of constructions necessary for installing these systems will be reduced correspondingly.

A modular principle of the NPP design is the most economically effective for reactors, in which the inherent safety properties against severe accidents have been realized to the maximal possible extent. First of all, this means the accidents with coolant's loss such as LOCA. To overcome these accidents, the LWRs need a lot of safety systems that are not necessary for RIs SVBR-75/100. This considerably reduces the construction scopes of the reactor compartment.

Control of the modular NSSS is carried out by an operator who uses the common power master unit. If there is any fault in the certain RI, it is automatically removed out of operation and can be cooled down autonomously with the turbine-installation systems.

A simple scheme of the RIs and similarity of their types allow to reduce the number of the operation and maintenance personnel at the modular NPP unit as compared with that at the NPP unit with one large-power RI with lots of safety systems including protection systems of localizing the accidents, control and providing systems. For example, the safety systems of the AP-1000 reactor have 184 pumps, 1400 driver valves, 40 km of the pipelines and cables [9].

A modular design of the NSSS power unit makes it possible to provide LF to be not less than 90 % under long reactor operation without refuelling. When each RI is shut down for refuelling, power unit's power reduces slightly.

Once-moment sequential refuellings of each RI included into the NSSS are equivalent to the mode of partial refuellings of the large-power reactor (1600 MWe) at annual refuellings of ~ 1/8 share of fuel each year). Duration and periodicity of scheduled maintenance and repair works are determined by requirements to the turbine-installation equipment.

Licensing of constructing the modular type large power power-unit will be much simplified in the case of constructing one RI or the small power modular power-unit which RI has been certified. Small power of the RI determines a comparatively low cost of its construction.

A plan and a longitudinal section of the SVBR-1600 reactor compartment's main building with the NSSS are shown in Fig. 6. The basic technical and economical parameters of the two-unit NPP based on RI SVBR-75/100 in comparison with those of the two-unit NPPs with RI VVER-1500, RI VVER-1000 (V-392), RI BN-1800 and HPP with ten steam-gas units PGU-325 are summarized in Table II [3].

Table II. Comparable parameters of different power plants

Name and Dimensions of the Parameter	NPP with RI SVBR-75/100	NPP with RI VVER-1500	NPP with RI BN-1800 [10]	NPP with RI VVER-1000	HPP with PGU-325
1. Set up power of the power-unit, (MWe)	1625	1479	1780	1068	325
2. The number of the units at the plant	2	2	2	2	10
3. Electric power necessary for plant's own needs, (%)	4.5	5.7	4.6	6.43	4.5
4. Efficiency of the net plant (power unit), (%)	34.6	33.3	43.6	33.3	44.4
5. Specific capital investments in the industrial construction of the plant, ($/kW)	661.5	749.8	783.4	819.3	600
6. Design cost of produced electricity, (cent/kW·h)	1.46	1.85	1.56	2.02	1.75

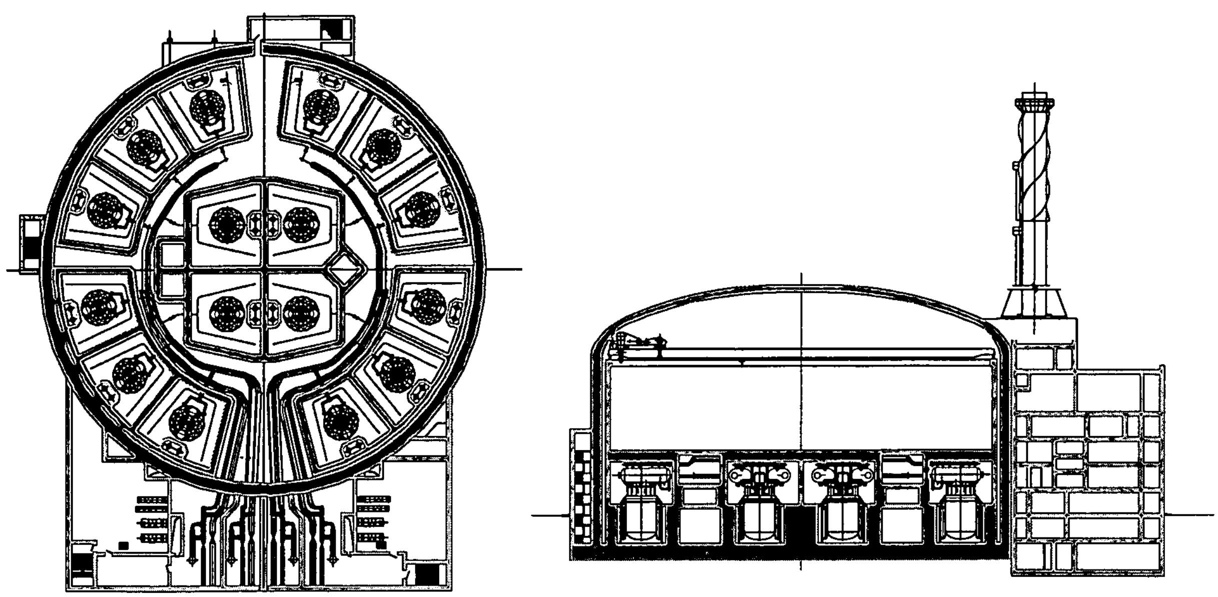

FIG. 6. A plan and a longitudinal section of the SVBR-1600 reactor compartment's main building with the NSSS

The results of technical and economical computations have revealed that in compliance with the data obtained at the conceptual design stage, the technical and economical parameters of the NPP with two 1600 MWe units, each based on the SVBR-75/100 type RI, are better than those of the NPP based on the large power LWRs and than those of the HPP with ten units PGU-325 operating by using natural gas. The term of constructing this NPP can be ~ 3.5 years.

However, it should be highlighted that reliability of the sited economical parameters of LWRs, which have been developing during several generations, is higher than those of the NPPs with RIs SVBR-75/100 because they have not had experience of practical realization. For that reason, the costs of the capital investments in the industrial construction of the NPP based on RIs SVBR-75/100 has an additional margin of 17 % over the standard one (60 % of the RI equipment cost). Besides, this NPP project that is actually the first generation design based on the conservative approach has a great potential for its development.

In the future, after carrying out corresponding R&D, the following technical solutions would improve the RI design and considerably improve the technical and economical parameters:

Use of the one-through SG producing super-heated steam that makes it possible to increase an efficiency of the thermo-dynamic cycle and increase electric power by 10 ... 15 %;

Increasing the LBC temperature at the reactor outlet at increasing the maximal temperature of the fuel elements' cladding from 600 to 650 °C that provides increasing the reactor heat power by 15 ... 20 % without changing its design.

Realization of these measures will make it possible to reduce the specific capital costs of plant's industrial construction to 560 $/kW and probably to a lower value.

6. FUEL CYCLE

Due to the low current costs of uranium and its enrichment, use of oxide uranium fuel with postponed reprocessing and SNF storing on the NPP site is economically justified for RI SVBR-75/100. Duration of this stage depends on the available resources of cheap uranium and NP scales. In compliance with the [11] data, in Russia an estimated term of expiring the cheap uranium resources will be 70 years at an average level of NPPs' total power being 45 GWe and in the world this value will be 40 years at an average level of NPPs' total power being 750 GWe.

However, because the increase of natural gas costs will overtake the increase of uranium costs, competitiveness of the NP will be assured even at higher costs of uranium caused by the fuel component's lower share in the electricity cost of the NPPs as compared with that of the HPPs.

At this stage the major way of improving the economic parameters of the fuel cycle will be increasing the lifetime duration (fuel burn-up depth) as experience in the core elements operation ability is gained and use of vibro-dense oxide fuel allowing deeper burn-up.

Further, an economically expedient will be the stage at which the own SNF will be reprocessed, NFC will be uranium closed (at adding enriched uranium into the NFC), plutonium, MA, fission fractions will be extracted and then stored.

Duration of the uranium stage can increase when changing over to the denser nitride fuel. At the same time, it could be expedient to only use uranium nitride for export in order to reduce the risk of unauthorized fissile materials proliferation at expanding the refuelling interval to only 15 years.

Actually, in the future it will be necessary to change over to the entirely closed NFC. The time period required for this change will be determined by appearing the developed in an industrial scale technology of SNF reprocessing that will be acceptable from the standpoint of RAW minimization and fissile materials non-proliferation. The existing technology of radiochemical reprocessing SNF do not meet these requirements. Besides, it will be only economically justified under stable operation of a large radiochemical factory (i.e. at reprocessing scales being 1000 ... 1500 t of SNF per year that corresponds to the total level of NPP set up power being ~60 ... 90 GWe).

One of the economically expedient variants of changing over to the entirely closed NFC that meets the necessary requirements is a technology based on using the "dry" methods of reprocessing SNF and a vibro-pack technology when the fuel elements are fabricated.

SSC RF-NIIAR have carried out the researches revealing that construction of power capacities on reprocessing SNF of SVBR-75/100 reactors and fresh FSA fabricating increases the specific capital costs of constructing the NPP by not more than 10 ... 15 % (about $ 76 /kW of set up power). And it has been presumed that reprocessing is performed on the basis of the pyro-chemical processes in the chloride melts and the reprocessing rate is 120 t of heavy metal per year (the highlighted reprocessing rate corresponds to the total NPP power on the basis of RIs SVBR-75/100 being ~ 12 GWe).

Change over to the closed NFC for the SVBR-75/100 reactors will have the lower cost if for fabricating the first fuel load from MOX fuel we use plutonium that has not been extracted from LWRs' SNF but has been extracted from the own SNF of uranium loads due to considerably lower scopes of reprocessing in terms of 1 t of plutonium. The quantity of plutonium extracted from SNF of three uranium cores is enough for fabricating one core from MOX fuel.

When reactors SVBR-75/100 operate in the closed NFC, economically effective use of LWRs' SNF as make up fuel without separation of uranium, plutonium, MA, and fission fractions instead of waste pile uranium is possible (similarly to the DUPIC-technology for the CANDU-reactors). That is, instead of reprocessing SNF of thermal reactors (both from the VVER and RBMK reactors) for the purpose to only extract 1 % of plutonium, after long storing during ~50 years this kind of SNF will be step by step utilized in the FR.

Due to the fact that the fraction of LWRs' SNF in fresh fuel of SVBR-75/100 operating in the closed NFC is ~ 10 ... 12 % and the plutonium fraction in LWRs' SNF does not exceed 1 %, influence of the plutonium isotopic vector in LWRs' SNF on the isotopic vector of fresh fuel is negligible for SVBR-75/100. Therefore, RI SVBR-75/100 makes it possible to develop a principally new strategy of the closed NFC that does not require expensive reprocessing SNF of thermal reactors for the purpose to extract plutonium for FRs' fuel supplying.

Flexibility of RI SVBR-75/100 relative to the fuel cycle technologies that is realized in compliance with a principle: "To operate using the type of fuel that is the most effective" makes it possible to postpone a task of constructing a specialized enterprise to several decades after the first unit of the NPP with such reactors has been put in operation. For example, after introduction of about 10 GWe of power capacities on the basis of RIs SVBR-75/100 and

repaying the costs of NPP construction, a certain share of profit could be spent on developing the industry on SNF reprocessing and fabricating FSA from MOX fuel.

After launching that factory, the cost of the core would be only determined by current operating costs of SNF reprocessing and FSA fabricating. If the SSC RF-NIIAR designs are used as a basis of that complex, contribution of fuel costs to the cost of the SVBR-75/100 core would be even less than that of the basic variant using oxide uranium fuel. This will make it possible to considerably improve the NPP competitiveness. This approach to construction of power capacities on SNF reprocessing and FSA fabricating presumes the owner of the NPP units is also the owner of the fuel cycle enterprise.

Along with this, on considering the prospects of economically substantiated closing the NFC of FRs caused by lack of uranium, some experts presume it might be required in 50 ... 100 years [12].

In this case it has been accounted that the cost of natural uranium is a very small fraction in the cost of electricity production. Even the opportunity of extracting uranium from the sea water (that seems exotic today) has been assessed and that cost will be 500 $ per kg.

7. LONG-LIVED RAW MANAGEMENT AND ENVIRONMENTAL IMPACT

In the course of NPP operation, liquid RAW is produced in very low quantities. This fact has been verified by experience of operating LBC cooled NSs' RIs.

The NPP design provides an installation for concentrating and solidifying the low quantities of liquid RAW.

After expiring the RI lifetime, the radioactive LBC can be many times recycled in the new RIs. In 1000 years of irradiation, slight residual long-lived radioactivity of LBC caused by Bi-208 and Bi-210m radionuclides will be lower than natural radioactivity of the uranium ore (in terms of U_3O_8). It will be only important at the final stage of NP functioning.

In this connection, LBC in the form of solid radioactive waste being disposed in the deep geological formations will not disturb the natural radioactivity equilibrium. Low chemical activity of lead and bismuth rules out radioactivity release into the biosphere. Therefore, the radio-ecological consequences of this disposal will be of no risk for the population of the next generations.

There is a similar problem for the LWRs as long-lived radionuclide zirconium-93 is forming in the fuel elements' zirconium claddings and channels.

The quantity of tritium release into the environment due to unavoidable water losses in the RI secondary circuit does not exceed ~ 50 TBq/GWe*yr that is within the limits of normalized release of tritium with liquid wastes into the environment of the world's operating NPPs.

Radioactivity release into the environment out of unloaded SNF is eliminated by a multi-barrier shielding against activity release out of the spent FSA. Being unloaded out of the reactor they are installed into the steel capsules filled with liquid lead. After lead solidifying four barriers are formed in the capsules: the fuel matrix, fuel element cladding, solidified lead, capsule vessel.

When operating in the closed NFC, fission products management does not presume their transmutation because of the low efficiency of the process.

Taking into account that the half life of majority of fission products does not exceed 30 years (except for technetium-99, iodine-129, cesium-135, and some others), it is supposed that after vitrifying they are placed into the "dry" control storage for about 300 years of long storing. After that storing, their activity will be determined by long-lived nuclides of technetium, iodine and cesium.

It is proposed to dispose these vitrified fission products in the deep geological formations with providing a multi-barrier shielding. (Instead of vitrifying a "sinrock"-technology may be used after verifying its advantages.) That method of fission products management rules out radioactivity release into the environment.

Management of transuranium (TRU) elements presumes that their release beyond the fuel cycle will be eliminated (except for very low losses at the stage of RAW chemical reprocessing) as they are well fissionable in a hard neutron spectrum of FRs and their concentration achieves a saturation condition very quickly.

To estimate the environmental impact caused by the NFC of SVBR-75/100, a value of specific radiotoxicity of transuranium elements (neptunium, plutonium, americium and curium) and long-lived fission products (technetium-99, iodine-129 and cesium-135) as a function of produced electricity was taken as a criterion.

The radiotoxicity standard was adopted as a volume of water necessary for diluting a built-up quantity of radionuclides to decrease its concentration down to the level meeting sanitary requirements to the drinking water in terms of specific radioactivity. Specific radiotoxicity is determined as SNF radiotoxicity divided by produced energy.

The following assumptions were made to evaluate radiotoxicity:

- MOX fuel with plutonium extracted from LWRs' SNF was used as a first load of the reactor;
- At the end of each lifetime and three-year cooling, SNF was reprocessed;
- Radiotoxicity of the main bulk of fission products with half-lives being less than 30 years was not accounted as after 300 years of cooling their radiotoxicity would be very low;
- Curium was extracted and transported to the temporary storage (repository) for 100-150-year cooling. After cooling, all radioactive isotopes of curium (except for curium-245) were transformed into plutonium isotopes. Then this isotopic mixture was transported back to the reactor for its further incineration [13];
- Mixture of plutonium, neptunium and americium with the rest uranium and necessary addition of depleted (waste pile) uranium was used for fabricating the fuel load for the next lifetime.

Figure 7 presents SNF long-lived specific radiotoxicity as a function of produced energy for reactor SVBR-75/100 within the NFC. Specific radiotoxicity of technetium-99, iodine-129 and cesium-135 in the final disposal is 0.014 km^3 /GWe*year that is nearly equal to that of natural uranium annually added to the fuel cycle in terms of GWe*year.

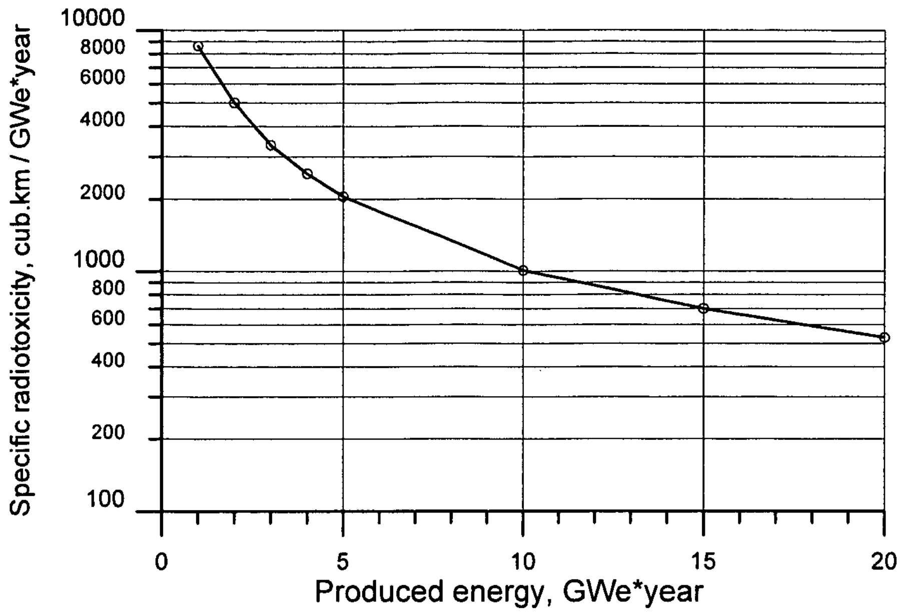

FIG. 7. SNF long-lived specific radiotoxicity.

The analysis of the obtained results shows the environmental-"friendly" effect of the NFC of SVBR-75/100 as specific radiotoxicity of long-lived RAW decreases at increasing the value of cumulative produced energy to the value of specific radiotoxicity of the extracted uranium ore. This is caused by the fact that the hard neutron spectrum in the reactor facilitates efficient incineration of both own MA, and MA built up in the LWRs.

8. TECHNOLOGICAL MAINTENANCE OF NON-PROLIFERATION

Non-proliferation of fissile materials means creating the conditions when inappropriate use of fissile materials is least attractive for potential distributors of the nuclear weapon.

It is evident that the problem of non-proliferation cannot be only solved by technological measures as despite the development of a new nuclear technology, there are the opportunities for illegal receiving the weapon materials and using the well-developed technologies of isotopic uranium separation and plutonium extraction out of spent fuel. For that reason, the complete solution to the problem of non-proliferation can be only achieved by coupling the technological and political measures.

Relationship of these measures will be different for nuclear and non-nuclear countries. During the recent decades all nuclear countries, which legally possessed the nuclear weapon, have solved this problem successfully using the measures of physical protection, accounting, control and safeguard. For that reason, the additional measures of technological maintenance of non-proliferation will be justified in case they do not reduce the NP competitiveness.

When using the NPP in developing countries, the additional measures of technological maintenance of NFM non-proliferation should be taken along with the political measures and international control.

In case of the export deliveries to the developing countries, the reactor design must eliminate access to the fuel. A reactor seller should keep the property rights to the reactor and core and provide necessary maintenance caused by periodical replacement of the core/reactor. A reactor module should be designed as a wholly replaceable unit after expiring the core lifetime (the goal is 15 years). This approach assures that the User-country should not possess the refuelling equipment, or the spent fuel repositories.

It is expedient to concentrate SNF reprocessing at the certain "closed" factories in nuclear or developed countries.

In the far future, when FRs are used world-wide, it would be possible to give up gradually the technologies of uranium enrichment and pure plutonium extraction for NP needs.

To reduce the risk of NFM unauthorized proliferation, the following measures realized at the different stages of the RI SVBR-75/100 lifetime including the NFC are considered below:

- The RI design eliminates uranium and thorium blankets that make it possible to accumulate fissile materials for the weapon purposes.
- Refuelling is performed very seldom (once a 7 ... 10-year period) and can be inspected easily. Partial refuelling is impossible.
- Refuellling is only performed by using the special equipment kit that is not delivered to the User-country.
- At the stage of fabricating the initial fuel load by using oxide uranium fuel, NFM non-proliferation is ensured by the core design due to use of uranium with U-235 enrichment being less than 20 %. This satisfies the IAEA requirements and allows to use these reactors in non-nuclear countries.
- At the stage of SNF storing, proliferation resistance is ensured by the fact that built-up plutonium together with high radiotoxic fission products is in the SNF ("the spent fuel standard"). Therefore, the possibility to steal SNF is eliminated and the SNF movements can be easily inspected by gamma-radiation.
- At the stage of SNF reprocessing, proliferation resistance is ensured by the fact that during technological reprocessing, built-up plutonium along with built-up MA is separated from uranium at non-deep purification from fission products. Therefore, plutonium stealing is impeded, and its applicability for fabricating the explosion devices becomes insignificant.
- At the stage of fabricating and transporting the MOX fuel, proliferation resistance is ensured by the fact that during fabricating re-fabricated fuel, 2 % of fission products built-up in SNF and all MA remain in it. This requires remote management of that fuel, impedes its stealing and facilitates the inspection of its movements.
- This fuel can be delivered to any countries as fuel management is only possible by using the special large heavy equipment that facilitates the accounting and inspection of fresh fuel.
- Fuel transportation in the reactor monoblock with solidified LBC creates an additional technical barrier to the fuel thefts.
- The IAEA inspection is ensured at all stages of the NFC.
- The measures of physical protection and safeguard are used.

9. POSSIBLE AREAS OF USING RI SVBR-75/100

High technical and economical parameters of RI SVBR-75/100, ability of the reactor monoblocks to be transported by railway, inherent safety properties of the RIs make conditions for their multi-purpose usage when they have been produced in large quantities.

First of all, it is renovation of the NPP units with LWRs which RIs have expired their lifetime. They can be renovated by installing the necessary number of RIs SVBR-75/100 in the empty SG and MCP compartments.

Results of technical and economical researches into technical opportunity and economical expediency of renovating the 2-nd, 3-rd, and 4-th Novovoronezh NPP (NVNPP) units on the basis of RIs SVBR-75 have revealed that renovation reduces two times the specific capital costs as compared with construction of the new replacing power capacities [14].

A similar renovation technology can be used for almost all LWRs units. In this case, the capital costs saving will be \$ 500 M per GWe (in Russian conditions) as compared with construction of the new replacing power capacities.

Experience gained by operating an industrial prototype of RIs SVBR-75/100 in conditions of the NVNPP will make it possible with a minimal investment risk to launch sequential renovation of the LWR units, which RIs have expired their lifetime and construction of modular NPPs with large power units in the countries, which power systems have high-voltage electric transfer lines with high transmission.

Taking into account the high extent of inherent safety of these RIs, it is expedient to use them for the heat supply needs.

In Russia: these are the regional NHPPs of 200 ... 600 MWe, which are necessary to be located near the cities. The term of constructing the regional NHPPs and their total cost will be much less than those of large power NPPs. Their construction can be realized at the expense of the finance sources of the Russian Federation subjects, including the joint stock but this require for the legislation to be changed.

Abroad: these are the power-complexes designed for producing electricity, heat and water desalination. Carried out by IAEA marketing studies have revealed that in developing countries that have no powerful electricity-transfer lines, there is a vast market for small power reactors of ~ 100 MWe.

Export potentials can be realized by granting on lease the transportable reactor unit for steam-supplying these power-complexes. In this case, the Supplier keeps the property rights, the Consumer needn't develop and maintain the complex infrastructure of fuel management, the Supplier takes all the possible risks.

In this case, the requirements to fissile materials non-proliferation are ensured by using uranium enriched in low than 20 %, lack of refuellings in the User-country. For refuelling the reactor unit should be transported to the User-country (once a 10-year period) in a nuclear and radiation-safe state due to the "freezing" LBC and core in the monoblock vessel.

10. CONCLUSION

- The inherent safety properties of RI SVBR-75/100, economic competitiveness of the NPPs with their usage, opportunity to operate in the closed NFC in the fuel self-providing mode or with low breeding, opportunity both to burn own MA in the reactor and use LWR's SNF as make up fuel, providing technological maintenance of non-proliferation make it possible to consider the proposed reactor technology as one of the most perspective trends in NP development.
- RI SVBR-75/100 meeting most of the requirements to the Generation IV DOE reactor systems and IAEA Project INPRO can be proposed as one of the basic installations of the collaborative International Project.
- The obtained results have revealed the technical opportunity and economical expediency of using the RIs of the SVBR-75/100 type for finding the solution to the certain basic tasks of Russian NP both in the nearest and far future at minimal launching costs of industrial mastering. First of all, this refers to the opportunity of economical effective considerable extending the NPP units' lifetime (by 30 ... 40 years) by renovating them.
- Use of the modular structure of the power-unit's NSSS makes credible an opportunity to change over in the future to the innovative technologies of a standard design of various capacity power units on the basis of the standard modules produced in quantities and a conveyer method of carrying out assembling works. This will make it possible to considerably reduce the terms of NPP construction and use a service base for technical maintenance of the reactor modules. The maintenance personnel will be also considerably reduced.
- Nuclear power based on the considered type NPPs can compete with heat power based on the modern steam-gas HPPs not only at the liberalized power market but at the investment market that will ensure the necessary pace of its development.
- To realize the highlighted above potentials, it is expedient and substantiated to construct the first RI SVBR-75/100 (an industrial prototype) as a part of the NPP unit by the year 2010. In Russian conditions the least cost of it will be if the proposed RI has been installed in the building of the shut down NVNPP second unit with using the existing constructions and some equipment. Carried out estimations have revealed that it will cost ~ $ 100 M.

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